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Preconceptual Engineering Services For The Next Generation Nuclear Plant (NGNP) With Hydrogen Production

NGNP Umbrella Technology Development Plan

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ACRONYMS AND ABBREVIATIONS

ADUN	acid-deficient uranyl nitrate
AFCI	Advanced Fuel Cycle Initiative
AGR	Advanced Gas Reactor
AGR Plan	Advanced Gas Reactor Fuel Development and Qualification Plan
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASR	area-specific resistance
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor [German pebble-bed HTR]
BDBA	beyond design basis accident
BISO	Coated-fuel particle design with two materials in coating system (low-density PyC and high-density PyC)
BSF	bench-scale facility
BWXT	BWX Technologies (formerly B&W)
CB	catcher bearing
CEA	Commissariat à l'Energie Atomique [French Atomic Energy Commission]
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CPS	control and protection system
CVD	chemical vapor deposition
DBA	design basis accident
DOE	US Department of Energy
DDN	Design Data Need
DGS	dry gas seal
DTF	"designed-to-fail" [fuel particles]
DV&S	design verification and support
EAB	exclusion area boundary
ED&D	engineering development and demonstration

EDP	engineering development plan
EFPD	effective full power days
EMB	electromagnetic bearing
EOL	end-of-life
EPA	[US] Environmental Protection Agency
EPACT	Energy Policy Act [of 2005]
EPRI	Electric Power Research Institute
EU	European Union
FDDM/F	Fuel Design Data Manual, Issue F
FFF	Fuel Fabrication Facility
FH&SS	Fuel Handling and Storage System
FOAK	first-of-a-kind
FP	fission product
FRG	Federal Republic of Germany
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain [HTGR]
FTC	Fuel Transfer Cask
FTE	fuel test element
GA	General Atomics
GT-MHR	[commercial] Gas Turbine-Modular Helium Reactor
GCR	gas-cooled reactor
GDDM	Graphite Design Data Manual
GNEP	Global Nuclear Energy Partnership
HEU	high-enriched uranium (usually ~93% U-235)
HFR	High Flux Reactor [Petten, NL]
HFIR	High Flux Isotope Reactor [ORNL]
HPC	high-pressure compressor
HPS	helium purification system
HTE	high temperature electrolysis
HTI	high-temperature insulation
HTIV	high-temperature isolation valve

HTGL	High-Temperature Gas Loop
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor [German pebble-bed HTGR]
HTTR	[Japanese] High Temperature Engineering Test Reactor
HX	heat exchanger
H2-MHR	Hydrogen [Production]-Modular Helium Reactor
I-NERI	International Nuclear Energy Research Initiative
IAEA	International Atomic Energy Agency
IFMU	In-Core Flux Mapping Unit
IMGA	Irradiated Microsphere Gamma Analyzer
IHX	intermediate heat exchanger
ILS	integrated laboratory-scale
IMF	inert matrix fuels
IMFPL	Initial Modular Fuel Process Line
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INET	Institute of Nuclear Energy Technology [China]
INL	Idaho National Laboratory
ISTC	International Science & Technology Center
ITRG	Independent Technology Review Group
IVM	“in-vessel metalwork” [ducting within PCU]
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
KFA	Kernforschungsanlage – Juelich [now renamed Forschungszentrum Juelich]
KIER	Korea Institute of Energy Research
KIST	Korea Institute of Science and Technology
LEU	low-enriched uranium ($\leq$19.9% U-235)
LMTD	logarithmic mean temperature difference
LPC	low-pressure compressor
LSM	Strontium-doped Lanthanum Manganite
LTR	Licensing Topical Report

MHR	Modular Helium Reactor
MHTGR	Modular High Temperature Gas-cooled Reactor (steam cycle)
MTR	materials test reactor
NCS	Neutron Control System
NDE	non-destructive examination
NDTT	nil-ductility transition temperature
NFI	Nuclear Fuel Industries
NGNP	Next Generation Nuclear Plant
NHDD	Nuclear Hydrogen Development and Demonstration
NHI	[DOE] Nuclear Hydrogen Initiative
Ni-YSZ	a mixture of metallic Nickel and Yttria-Stabilized Zirconia
NIIAR	[Scientific Research Institute for Atomic Reactors (Also known as the All-Russian Institute of Atomic Reactors RIAR), Dimitrovgrad, RF]
NP-MHTGR	New Production-Modular Helium Reactor
NRC	[US] Nuclear Regulatory Commission
NRR	Neutron Radiography Reactor
O&M	operation and maintenance
OCV	open-cell voltage
OKBM	[Experimental Bureau of Mechanical Engineering, Russian]
ORNL	Oak Ridge National Laboratory
PAG	Protective Action Guide
PB	Peach Bottom
PBMR	Pebble Bed Modular Reactor [South Africa]
PCD&IS	Plant Control, Data and Instrumentation System
PCDSR	preconceptual design studies report
PCHE	printed circuit heat exchanger
PCS	power conversion system [GA design]
PCU	power conversion unit [OKBM design]
PCV	power conversion vessel
PHTS	Primary Heat Transport System
PHX	process heat exchanger
PIE	postirradiation examination

PIH	postirradiation heating
PIRT	Phenomenon Identification and Ranking Table
PPMP	[NGNP] Preliminary Project Management Plan
PRA	probabilistic risk assessment
PSER	Preliminary Safety Evaluation Report
PSID	Preliminary Safety Information Document
PyC	pyrocarbon [coating]
QC	quality control
R&D	research and development
R/B	release rate-to-birth rate ratio [a metric for fission gas release]
RCCS	reactor cavity cooling system
RF	Russian Federation
RN	radionuclide
ROK	Republic of Korea
RPS	Reactor Protection System
RPV	reactor pressure vessel
RS	Reactor System
RSM	rotor scale model
RTDP	Regulatory Technology Development Plan
SCS	Shutdown Cooling System
SHTS	Secondary Heat Transport System
SI	sulfur-iodine [process for thermo chemical water splitting]
SOEC	solid oxide electrolyzer cell
SOFC	solid oxide fuel cell
SMR	steam-methane reforming
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SR	shear ratio
SRM	System Requirements Manual
SS	stator seal
SSC	systems structures and components
STP	standard temperature and pressure [0 ^{o}C (273 K) and 1 atm]

TBD	to be determined
TC	turbocompressor
TCSS	turbocompressor stator seal
TDP	technology development plan
TDPP	Technology Demonstration Program Plan [Russian]
TM	turbomachine
TRISO	Coated-fuel particle design with three materials in coating system (low-density PyC, high-density PyC, and SiC)
UCO	uranium oxycarbide - an admixture of UC_2 and UO_2
USDOE	US Department of Energy
V&V	verification and validation
VHTR	Very High Temperature Reactor
VLPC	vented low pressure containment
VS	Vessel System
WPu	weapons-grade plutonium
XV	cross vessel

1. SUMMARY

The primary purpose of this technology development plan (TDP) is to focus and prioritize the R&D programs needed to support the Next Generation Nuclear Plant (NGNP) Project. The first step to that end was to specify an NGNP "reference" preconceptual design, which is documented in the Preconceptual Design Studies Report (PCDSR 2007), because many Design Data Needs (DDNs) are design-specific (other DDNs such as those related to fuel and fission products are largely generic). The status of the various technologies needed to support NGNP design and licensing was then reviewed, and DDNs were defined where the current database was judged to be inadequate. Many of the resulting NGNP DDNs, especially those related to the Reactor System and Power Conversion System, are the same or similar to the commercial GT-MHR DDNs, but a number of new DDNs were also identified, especially related to the intermediate heat exchanger (IHX) and the hydrogen production processes.

The DOE-sponsored technology programs intended to support the NGNP, including the various NGNP R&D programs and the DOE Nuclear Hydrogen Initiative (NHI) programs, were then evaluated, and their responsiveness to the DDNs was assessed. The existing TDPs were critiqued on an exception basis to identify any deficiencies and unnecessary workscope, especially in the context of the NGNP schedule. In general, these TDPs propose to investigate an excessively large number of materials (graphite, metals, etc.); consequently, the candidate materials need to be prioritized.

The results of the evaluation are summarized below and elaborated in the body of this TDP. The subject evaluation was made on the basis of the "reference" NGNP preconceptual design described in the PCDSR and the R&D program plans made available by INL and on various DOE web sites (e.g., the NHI 10-yr plan). When the NGNP reference design is officially declared and subsequently matures, additional DDNs will undoubtedly be defined, but it is anticipated that the major ones have been identified here.

Likewise, the GA team perception regarding the responsiveness of the NGNP and NHI R&D programs to the DDNs may change to some degree as those R&D plans are better defined. Overall, the current NGNP and NHI R&D plans appear largely adequate to meet the DDNs with a number of important caveats that are described below by technology area. However, with the notable exception of the AGR fuel development plan, these R&D plans are, in general, too high level and largely qualitative in nature (e.g., few test matrices, etc.). Consequently, a general recommendation is that the NGNP and NHI program plans be revised to tie them directly to the NGNP DDNs and that they be better quantified. Without more specificity, it is not clear what data will be available at what time, and it is not possible to judge the reasonableness of the R&D cost estimates.

1.1 "Reference" Preconceptual NGNP Design

The "reference" preconceptual NGNP design upon which this TDP is based is presented in (PCDSR 2007). The required plant design and licensing schedule is based upon planning Option 2 ("Balanced Risk") in the NGNP Preliminary Project Management Plan (2006) which requires a 2018 plant startup. In large measure, this preconceptual design is based upon the commercial GT-MHR conceptual design developed for DOE in the early 1990s with modifications for operation with an increased core outlet temperature of 950 ^{o}C and with the addition of a second loop to the primary circuit to supply process heat via an intermediate heat exchanger to hydrogen production plants utilizing both the sulfur-iodine (SI) thermochemical water-splitting process and the high temperature electrolysis (HTE) process. The selection of this "reference" design determines the DDNs and the GA team's evaluation of the responsiveness of the NGNP and NHI R&D plans to the DDNs. In particular, several key design selections determine much of the required technology development to support design and licensing; the major ones are described below.

1.1.1 Prismatic Core

The GA team strongly recommends a prismatic core for the NGNP for a large number of reasons elaborated in the PCDSR. While there are a number of DDNs, especially in the fuel, fission products, graphite and high temperature metals areas, that are generic to both prismatic and pebble-bed cores, many design verification and support (DV&S) DDNs related to the Reactor System are core design specific. The NGNP Reactor System DDNs are essentially the same as those for the commercial GT-MHR with appropriate modifications for the higher core outlet temperature.

1.1.2 Increased Core Outlet Temperature

The decision to increase the core outlet temperature from 850 ^{o}C for the GT-MHR to 950 ^{o}C for the NGNP, in order to increase the thermal efficiency of both electricity- and hydrogen production, introduces a number of materials challenges, especially for the high-temperature metals used for the intermediate heat exchanger, reactor pressure vessel (RPV), and turbine blades. The ultimate impact of the higher core outlet temperature will be strongly influenced by the extent to which design innovations can be applied to mitigate the negative effects of a higher gas temperature. For example, the addition of blade cooling may compensate for higher turbine inlet gas temperature but at the expense of somewhat reduced thermodynamic efficiency. Likewise, core optimization may accommodate the higher outlet temperature without an increase in peak fuel temperatures during normal operation. In general, the final implications for the R&D programs cannot be determined until near the end of Preliminary Design.[1] However, the need to qualify a high-temperature metal, such as IN 617, for the IHX will undoubtedly remain and represents a top-priority DDN for the NGNP.

1.1.3 Power Conversion System

The GA team recommends the direct Brayton-cycle Power Conversion System (PCS) developed by OKBM on the DOE/ROSATOM-sponsored International GT-MHR program for disposition of surplus weapons-grade plutonium (WPu) with modifications as necessary for operation with a turbine inlet temperature of 950 ^{o}C. It was recognized from the beginning that

[1] In this TDP, "Conceptual Design," "Preliminary Design," and "Final Design" will be capitalized when referring specifically to these design phases for the NGNP.

the vertical integrated PCS concept poses several technical challenges with respect to individual component designs and their arrangement within a single PCS vessel. On the other hand, it was also clear that there are substantial technical and economic incentives for such a selection. Given the technical challenges associated with the integrated PCS configuration, the design development was carefully monitored by the GT-MHR project through a series of design reviews, both by internal experts and by independent third party experts. The results of these technical reviews were thoroughly reviewed and evaluated to identify the uncertainties and unconfirmed assumptions (i.e., technical issues) in the science or engineering upon which the design is based. A series of DDNs and a technology development program plan were then prepared to develop the data needed to qualify the OKBM PCS design. Supplementary DDNs have been identified for 950 ^{o}C operation.

1.1.4 Hydrogen Production

The Energy Policy Act of 2005 mandates that the NGNP mission include hydrogen production. This legal mandate, which is perhaps the greatest incentive for the NGNP Project, is responsible for the considerable additional technology development required for the NGNP compared to the commercial GT-MHR. Hydrogen production necessitated the addition of the two hydrogen plants as well as the primary and secondary heat transfer loops, including the technically challenging IHX, and increased thermodynamic hydrogen production efficiency was the primary motivation for increasing the core outlet temperature as well. Both the SI and HTE processes for hydrogen production are immature technologies compared to gas-cooled reactor technology which has been under development internationally for more than four decades. The DDNs for SI and HTE are substantial and technically challenging, and it is expected that additional DDNs will be identified as the technology development progresses from the current laboratory-scale to pilot plant-scale to engineering-scale testing.

1.2 Technology Base for NGNP Design and Licensing

The status of several key elements of the technology base is described below. These particular technology areas were chosen because significant additional technology development is required in these areas for NGNP design and licensing. The status of the key technologies is elaborated in Section 3 which includes an extensive bibliography.

1.2.1 Nuclear Heat Source

The technology base for MHR design and licensing derives from five decades of international R&D programs combined with the design, construction and operation of seven He-cooled reactors. Actual reactor operation provides the most credible demonstration of the technology.

1.2.1.1 Radionuclide Containment

The radionuclide containment (RN) system for an MHR, which reflects a defense-in-depth philosophy, is comprised of multiple barriers to limit radionuclide release from the core to the environment to insignificant levels during normal operation and a spectrum of postulated accidents. The five principal release barriers are: (1) the fuel kernel, (2) the particle coatings, particularly the SiC coating, (3) the fuel element structural graphite, (4) the primary coolant pressure boundary, and (5) the reactor building/containment structure. The effectiveness of these individual barriers for containing radionuclides must be characterized for normal operation and a broad spectrum of postulated accidents.

The most important barrier in the RN containment system is the TRISO coating system. TRISO particle fuel has been fabricated in many countries throughout the world, irradiated in numerous irradiation test capsules, and used as the fuel in power and experimental reactors; thus, the basic processes for fabrication of fuel High Temperature Gas-cooled Reactors (HTGRs) are well established. However, the fuel quality requirements for future advanced HTGRs are considerably more stringent than for these earlier reactors. The capability of TRISO fuel particles to meet these stringent performance requirements has been demonstrated in Germany for the pebble-bed reactor design, but has not yet been demonstrated in the USA (or elsewhere) for prismatic core designs.

A radionuclide containment issue of special interest for the NGNP is containment of tritium. Tritium will be produced in a MHR by various nuclear reactions. Given its high mobility, especially at high temperatures, some tritium will permeate through the IHX and hydrogen plant process vessels, contaminating the product hydrogen. This tritium contamination will contribute to public and occupational radiation exposures; consequently, stringent limits on tritium contamination in the product hydrogen are anticipated to be imposed by regulatory authorities. Design options are available to effectively control tritium in the NGNP, but they can be expensive so an optimal combination of mitigating features must be implemented in the design.

1.2.1.2 High Temperature Materials

By definition, structural materials operate at high temperatures in High Temperature Gas-cooled Reactors where coolant temperatures during normal operation can be as high as 950 ^{o}C (and even higher during transients). The structural materials that experience the highest service temperatures are the core graphite and the high temperature metals used for the reactor internals and hot duct.

Graphite has been used as a moderator and a structural material for nuclear reactor cores since the dawn of the nuclear age. Certain graphite properties are of critical importance to the proper functioning of the core. For example, stringent limits were imposed upon primary coolant oxidants in Fort St. Vrain (FSV) because of concerns about oxidation of the PGX graphite core support floor which was aggravated by high iron impurities.

The design of the graphite components is based on a considerable international body of graphite data. In the early 1970's, a near-isotropic, petroleum coke-based graphite, designated Grade H-451, was developed by Great Lakes Carbon (now a part of SGL Carbon); numerous test programs and experiments were conducted to characterize its behavior. H-451 was used successfully in FSV reloads, and it was the reference fuel element graphite for the NP-MHTGR. Unfortunately, this graphite is no longer commercially available, and a priority task for the NGNP technology program is to identify and qualify a replacement graphite with comparable properties.

The component models and material property data for designing graphite components are documented and controlled in the GA Graphite Design Data Manual. These data will be used in the Conceptual Design (and perhaps Preliminary Design) of the NGNP core until a replacement graphite is characterized.

Structural metals are used throughout the primary coolant circuits of HTGRs, including the reactor internals and heat exchangers. When the first HTGRs were designed, it was obvious that the metallic components would operate at high temperature and that some would be exposed to high neutron doses as well. The environmental aspect that was not fully anticipated until the first prototype HTGRs were operated was the extent to which the reactor primary coolant chemistry could vary.

The design of the reactor metal components is based on the American Society of Mechanical Engineers (ASME) Code with conservative reductions in Code allowables based on existing data relative to environmental effects on the various alloys. Since the early 1960s, numerous test programs and experiments have been conducted in support of metals technology for HTGRs. Extensive laboratory testing, using a range of temperatures and helium impurity levels, has been carried out in the USA, Europe, and Japan over the past three decades to verify the performance of a variety of high-temperature materials in helium environments expected for HTGR systems. Test materials have included wrought alloys such as 2¼Cr-1Mo steel, Alloy 800H, Hastelloy X, Inconel 617 and other metals.

The greatest materials challenge for NGNP design will be to qualify a metal for the IHX which can operate at 950 ^{o}C with a long lifetime (IN 617 is the leading candidate). The Japanese HTTR has an IHX made of Hastelloy XR which operates at 950 ^{o}C with a 10-yr design lifetime.

1.2.1.3 Heat Transfer Technology

Historically, the heat transfer technology relied upon in gas-cooled reactors has typically been based on helical-coil heat exchanger technology. Helical-coil heat exchangers are the preferred design choice for steam generators and are also used in emergency core cooling system, auxiliary cooling systems, and shutdown cooling systems. Helical-coil heat exchangers often have a relatively large logarithmic mean temperature difference (LMTD) in order to reduce the heat transfer surface area and size of the heat exchanger.

The printed circuit heat exchanger (PCHE) technology achieves high effectiveness and low LMTD in a compact heat exchanger with reasonable pressure drops across the heat exchanger. PCHEs consist of alternating metallic plates in which microchannels have been chemically etched and then joined together under high pressure and temperatures to form a diffusion-bonded heat transfer core. PCHE technology has been applied to numerous industries but has yet to be applied in the nuclear industry – especially for gas-cooled reactors at the very high temperatures. A PCHE IHX is recommended for the NGNP with a helical-coil design as a backup.

1.2.1.4 Power Conversion System

Early versions of the MHR utilized a power conversion system based on the Rankine cycle (i.e., steam cycle), but a direct Brayton cycle was adopted as part of the design evolution that was driven by the need to make the MHR more economically competitive with other electricity generation options. The initial preconceptual GT-MHR design was developed under a joint initiative of the DOE and US utilities over the period 1991 - 1994. A vertical integrated power conversion system (PCS) design was selected from trade studies performed as part of the GT-MHR preconceptual design development.

The major components in the PCS are based on combustion gas turbines (both industrial and aeroderivative units) that are in service today for electrical power generation. The major components include the following: turbocompressor, magnetic bearings, electrical generator, recuperator, precooler/intercooler, and pressure vessel. The reference PCS design uniquely packages together the major components to achieve a highly efficient compact unit. The fact that the major components are based on proven hardware reduces development risk for the power conversion system. In the 1970s, two large helium turbine facilities were built and operated in Germany; the experience gained was factored into the design selections made for the reference PCS.

1.2.1.5 Design Verification and Support

The base technology for designing most MHR systems, structures and components (SSC) derives from five decades of international R&D programs combined with the design, construction and operation of seven He-cooled reactors. For the NGNP preconceptual design, the important exceptions are the PCS, IHX and hydrogen plants which are discussed separately.

1.2.1.6 Design Methods Development and Validation

The design methods for analyzing prismatic HTGRs were first developed to support the design and licensing of Fort St. Vrain and the large HTGRs in the 1970s. A brief summary status of the prismatic core design methods is presented below. Most of the design methods used for the analysis of the plant systems, structures and components are commercially available design tools, such as ANSYS, SINDA/FLUENT, RELAP5, Pro/E, etc., and they will not be addressed herein since there is a whole literature devoted to them.

GA's reactor physics codes were originally developed from basic neutron transport and diffusion theory. These methods were adapted to high-temperature, graphite-moderated systems to allow calculation of temperature-dependent graphite scattering kernels, and the development of fine group cross sections for graphite systems from point-wise data (e.g., ENDF/B, JEF, and JENDL data sets). These GA nuclear design methods have been benchmarked against other industry standard codes, such as MCNP, and integral test data from operating HTGRs and critical experiments with generally good agreement. While the experimental data used for nuclear code verification and validation (V&V) are considered reliable, some of the older data and, in particular, the international data may not have an adequate QA pedigree to be accepted by the NRC without some confirmatory testing.

The basic approach for performing core thermal/fluid flow analyses for prismatic HTGRs was also established to support the design of FSV and the large HTGRs in the 1970s, and a number of codes were written at GA for that purpose. While the analytical tools have evolved and the computational capabilities have improved enormously with modern computers, the basic analytical approach is still valid. Future core thermal/flow analysis for normal operation and accidents will be performed with industry standard codes, such as ANSYS and RELAP5, and various commercial CFD codes as required.

Design methods have also been developed to predict the various fuel performance and radionuclide transport phenomena in HTGRs in order to generate source terms for plant design and safety analysis. The accuracy of these design methods has been assessed by comparing code predictions with data from operating reactors and integral test data from various experimental programs. In general, the uncertainties in the predicted source terms are large. These design methods are adequate for predicting source terms during NGNP Conceptual Design, but they will need to be upgraded during Preliminary Design and validated prior to completion of Final Design.

A number of core structural analysis codes were developed at GA during the past three decades and used extensively for core design and safety analysis. However, future core structural analysis, including seismic analysis, will be performed with ANSYS and ANSYS/DYNA3D. Improved constitutive equations for graphite along with improved material property data will be required.

1.2.2 Hydrogen Production

The technology base for hydrogen production derives primarily from two sources: (1) the commercial production of inorganic chemicals for more than a century for the SI process, and (2) international development of solid-oxide fuel cells (SOFC) for the past three decades for the HTE process.

1.2.2.1 SI Process

The SI thermochemical water-splitting process was invented at GA in the early 1970s. The modern DOE-sponsored R&D effort on the SI process has been done primarily in collaboration with the French Commissariat à l'Energie Atomique (CEA) under an International Nuclear Energy Research Initiative (I-NERI) agreement since 2003. Throughout 2004 and 2005, experimental work in glass equipment was conducted to evaluate and choose appropriate methods for carrying out the reactions in each of the three sections of this process. Design work in 2006 allowed for lab-scale devices to be constructed in 2007 from engineering materials that are expected to be used in a pilot-scale hydrogen production facility scheduled for operation beginning in 2013.

The highly corrosive nature of some chemical streams in the SI process has led to significant research work in the area of materials compatibility. Early screenings showed that alloys of tantalum appeared to be suitable, and current work is exploring long-term performance and corrosion resistance of materials stressed or machined in ways that materials of construction for larger scale plants will experience.

1.2.2.2 HTE Process

The solid-oxide electrolyzer cell (SOEC), which is the fundamental component enabling the HTE process, is essentially a solid-oxide fuel cell operating in reverse. SOFCs have been under international development for more than two decades and appear to be approaching commercial viability for a number of applications.

SOEC concepts based on both planar-cell and tubular-cell configurations are currently being developed. SOEC technology based on the planar-cell concept is being developed as part of the DOE NHI program and involves collaboration between INL and Ceramatec of Salt Lake City, UT. A potential issue for the planar-cell concept is stack durability and sealing as the result of thermal cycling. Tubular cells have less active cell area per unit volume than planar cells but are less susceptible to these issues. Toshiba Corporation is currently developing an SOEC concept based on the tubular-cell configuration. The GA team, which includes Toshiba, concludes that both the planar-cell and tubular-cell configurations are promising technologies for future commercialization and recommends that both concepts be developed through at least the pilot-scale demonstration stage so that tradeoffs between capital costs and long-term performance can be accurately characterized.

1.3 Design Requirements

The design requirements imposed upon the NGNP are defined in the Systems Requirements Manual (SRM), and these requirements will ultimately determine what technology is needed to support plant design and licensing. Consequently, determination of the design requirements is a prerequisite to defining the Design Data Needs and attendant technology development programs for the NGNP. In fact, the current NGNP and NHI R&D programs lack focus because they are, in general, generic programs that have not been scoped or prioritized to support a particular plant design. Since the NGNP is still in the preconceptual design phase, the design requirements are provisional, especially the lower-level ones; consequently, the conclusions presented in this TDP regarding the current R&D programs are subject to revision as the design matures and more definitive feedback is provided by regulators and potential customers.

As described above, there is a large, often robust, international database to support most aspects of NGNP design as a result of five decades of nuclear power plant design and operation, especially the design and operation of seven HTGRs. Consequently, most design requirements do not generate DDNs and can be satisfied by standard engineering practice and by application of validated analytical tools. In fact, a relatively few design requirements generate most of the DDNs that have been identified for the NGNP at this time, and they in large measure drive the technology development requirements. These requirements are presented in Section 4.

1.4 Evaluation of the NGNP and NHI Technology Development Plans

The NGNP DDNs are summarized in Section 5, and the key DDNs for each technology area are listed in Table 1-1 (the choice of "key" DDNs was somewhat subjective at the preconceptual design phase). As already mentioned, the DDNs for the commercial GT-MHR apply almost without exception to the NGNP preconceptual design. A number of new DDNs have also been identified for the NGNP, largely because of its hydrogen production mission.

The DOE-sponsored technology programs intended to support the NGNP, including the various NGNP R&D programs and the NHI programs, were evaluated, and their responsiveness to the NGNP DDNs was assessed. The existing TDPs were critiqued on an exception basis to identify any deficiencies and unnecessary workscope, especially in the context of the NGNP schedule. The results are summarized in Table 1-1 and elaborated in Section 6.

1.4.1 Fuel/Fission Products Program

The DOE AGR Fuel Development and Qualification Program (AGR Plan/1) has the mission to develop and qualify fuel for the NGNP. The AGR fuel program is developing and qualifying conventional, SiC-based TRISO fuel particles with the assumption that conventional TRISO particles will be adequate for use in the initial core of the NGNP. However, there was no NGNP reference design when the AGR Fuel Program was first planned in 2003. Consequently, the program initially selected the GT-MHR fissile particle as the reference particle design for fuel fabrication process development and irradiation testing. Validation of radionuclide source terms is also within the scope of the AGR Fuel Program.

The AGR program plan is a comprehensive plan which the GA team continues to endorse with the caveats summarized below (GA was a member of the team who prepared the AGR plan and continues to participate in the program). The NGNP Preliminary Project Management Plan (PPMP) and the Independent Technology Review Group (ITRG) identified a number of risks associated with the overall NGNP fuel qualification effort. The GA team agrees with these concerns and has identified additional deficiencies. The scope of the AGR program is largely responsive to the NGNP DDNs; the fundamental problem is that the AGR program schedule does not support the NGNP design and licensing schedule. Moreover, given the limited existing test facilities in the USA, it would be difficult to significantly accelerate the AGR program even with unconstrained funding.

1.4.1.1 Fuel Process Development

The AGR fuel development schedule does not support a 2018 startup of the NGNP. As an expedient, GA proposes the use of 10%-enriched UO_2 TRISO fuel fabricated by Nuclear Fuel Industries (NFI) in Japan for the NGNP first core fuel load (and possibly for one or more reload segments). However, GA views this as a necessity only to allow the 2018 startup because the NGNP Project must develop a domestic supply of UCO TRISO fuel (assuming that the NGNP is a prismatic block MHR) in order to meet the NGNP project objectives as identified in the NGNP PPMP. Consequently, GA believes that it is essential that the NGNP Project build, license, and operate a fuel manufacturing pilot plant for the NGNP to demonstrate the viability of economical mass production of UCO TRISO fuel, thereby satisfying the fuel fabrication process DDNs

The 510 fuel element/year process line that would be built and demonstrated in the NGNP Fuel Fabrication Facility (FFF) during production of the second NGNP core would be the basic production module that would be replicated in a commercial fuel fabrication facility (comparable to the Initial Modular Fuel Process Line planned for the NP-MHTGR project). Thus, the NGNP FFF would demonstrate the fuel fabrication technology needed for a commercial fuel supply business, thereby greatly reducing the costs and risk that would be associated with a first-of-a-kind fuel manufacturing facility. This conclusion is, of course, based on the premise that the US government would make the NGNP pilot line technology available to any US company that wishes to replicate the technology to develop a commercial MHR fuel manufacturing business.

Another issue with respect to fuel process development is coater scale-up. The fuel currently being irradiated in AGR-1 was made in a laboratory scale coater at ORNL. Coating process development is currently proceeding at BWXT to scale up the coating process to a 15-cm diameter coater. Commercial scale coaters operated at GA and at HOBEG GmbH in Germany had a diameter of 24 cm. The AGR program recognizes the need to scale up the coating process to commercial coater, but the second scale-up step is not currently in the AGR program plan.

1.4.1.2 Fuel Materials Qualification

Both the ITRG and the PPMP have recognized the risks associated with the AGR Fuel Program's single-path approach to fuel qualification. Indeed, the PPMP calls for expansion of the program to include a dual path involving irradiation testing of UCO fuel fabricated in the USA by BWXT and UO_2 fuel fabricated by NFI. The UCO fuel would be irradiated in test AGR-2 as originally planned, and a new irradiation test ("AGR-2a") would be added to the program for irradiation testing of UO_2 fuel fabricated by NFI. Irradiated fuel from both irradiation tests would be subjected to heating tests to simulate accident conditions (i.e., "safety tests"). The irradiation and safety testing of NFI UO_2 fuel is not currently included in the AGR Plan.

GA endorses the approach described in the PPMP to irradiate both UCO fuel and NFI UO_2 fuel. However, consistent with GA's view that demonstration of UCO fuel in the NGNP is essential for deployment of commercial MHRs in the USA, GA does not agree that a down selection for qualification testing should be made between UCO fuel and NFI UO_2 fuel. Rather, UCO fuel should be qualified as rapidly as possible, and NFI UO_2 fuel should be qualified for use in the initial core and early reload(s), based on Japanese and confirmatory US irradiation and safety test data. It is also assumed that a fuel performance monitoring program in the NGNP would be necessary to supplement the irradiation and safety testing data for NFI UO_2 fuel.

1.4.1.3 Radionuclide Transport

As indicated in the PPMP, there is a substantial risk that the RN transport workscope included in the AGR Plan will be inadequate to support NGNP design and licensing. This problem has been exacerbated by chronic funding shortfalls for the AGR program; consequently, no experimental work in the RN transport area has been initiated to date with the exception that the driver fuel has been fabricated for irradiation tests AGR-3 and AGR-4. In fact, no experimental work on RN transport outside of the core is planned until FY12. The significant RN transport issues identified with the AGR Plan are summarized below.

A series of fission product transport tests in an in-pile loop are needed in order to generate the integral test data necessary to validate the predicted source terms for the NGNP. The AGR Plan contains tasks to construct an in-pile loop and to perform an in-pile test program. However, the design and construction of the loop are not initiated until FY13. The technical feasibility of constructing such a facility (presumably in the ATR at INL) and the attendant costs and schedule must be established far earlier if the design methods for predicting RN transport in the primary circuit are to be validated before the end of Final Design. In addition, the cost and schedule estimates for loop design and construction appear to be extremely optimistic.

The AGR Plan does not address tritium transport (perhaps, in part, because it is a generic development plan which does not focus on a specific reactor design). Tasks to characterize tritium retention in the core and tritium permeation through heat exchanger materials need to be added to address NGNP DDNs.

The AGR Plan does not address RN transport in the vented low-pressure containment (VLPC). It only includes an evaluation of the extent to which the experimental water-reactor database for RN transport in high-pressure containment buildings might be applicable to the VLPC. A recent evaluation concluded that these data are of limited value for refining and independently validating the design methods used to predict RN transport in VLPCs because the radionuclide concentrations and the physical and chemical forms in the two systems are too different. As a result, new DDNs have been identified that the AGR program needs to address.

1.4.2 Structural Materials R&D Program

The objective of the NGNP Materials R&D Program (2005) is to provide the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant (which is included in the NHI program). The most important products of the program will be qualified nuclear graphite for the reactor core and high temperature metals for use throughout the nuclear heat source, power conversion system, primary heat transport systems, and balance of plant. The GA team perspective on the graphite and metals R&D program is briefly summarized below.

1.4.2.1 Graphite Program

The graphite program described in the NGNP Materials R&D Program Plan is evaluating at least 16 nuclear graphites and fuel-element matrix materials from at least four international graphite vendors. The current focus of the program is the graphite irradiation capsule AGC-1 which is intended to provide irradiation creep- and dimensional change data on candidate graphites for the use in the NGNP. Creep data will be obtained for six major graphite grades (vendor in parenthesis): H-451 (SGL) and IG-110 (Toyo Tanso), both of which are included as reference graphites, and four new grades, PCEA (Graftech), NBG-17 (SGL), NBG-18 (SGL), and IG-430 (Toyo Tanso). In addition, AGC-1 contains ten minor grades of graphite.

A comprehensive, stand-alone graphite TDP is urgently needed which defines the entire scope, schedule and cost of the planned program. The planned program is probably responsive to the graphite DDNs defined herein for a prismatic NGNP, but it may be excessive. The graphite service conditions in a prismatic VHTR are not demanding (e.g., fast neutron fluence to the fuel-element graphite is <5 x 10^{21} n/cm^2, E >0.18 Mev). Previously qualified H-451 for fuel and reflector elements and Stackpole 2020 for the core support structure have adequate material properties.

From the GA team's perspective, the primary requirement for the NGNP Project is to identify and qualify a replacement graphite for H-451. The recommended approach is to use AGC-1 as a screening capsule to identify the lowest-cost graphites with properties comparable to H-451 and then to perform supplemental testing to establish a correspondence between the behavior of the replacement graphite and the extensive H-451 experience base. While it is important to minimize the number of graphites to be characterized, two or more domestic suppliers of H-451 replacement graphite should be qualified. The GA team considers the qualification of a replacement graphite for H-451 to be a high priority, but a low risk, task.

1.4.2.2 High Temperature Metals

The metals program described in NGNP Materials R&D Program Plan is evaluating a large number of alloys for high temperature applications throughout the Reactor System, Power Conversion System, and Primary- and Secondary Heat Transport Systems.

With an important exception, planned program appears responsive to the structural metals DDNs defined herein for a prismatic NGNP, but it may be excessive from the GA team's perspective. Since the reference NGNP design has not been chosen, the current materials R&D program is necessarily a generic program. Once the reference design is determined, the metals R&D program needs to be focused on a relatively few alloys (e.g., a prime and a backup alloy for each application). To that end, a comprehensive, stand-alone metals TDP should be prepared which defines the entire scope (test matrices, etc.), schedule and cost of the planned program. A high-priority task will undoubtedly be to complete qualification of IN 617 for an IHX operating at 950 ^{o}C.

An important deficiency in the current metals R&D program is that it does not include turbine blade alloys (e.g., IN 100, IN 738). There is considerable incentive to develop and qualify a turbine-blade alloy which could be used at 950 ^{o}C without blade cooling with an acceptable service lifetime. Blade cooling is a viable alternative, but the thermodynamic efficiency will be somewhat reduced. Thermal barrier coatings may be needed as well. The turbine blade alloy R&D program should emphasize helium effects as well as thermal fatigue, and the threshold concentrations and temperatures for possible corrosion of turbine alloys by radionuclide plateout (Te, Cs, Ag) should be investigated.

1.4.3 Energy Transfer Technology Program

The GA team understands that an Energy Transfer TDP will be prepared in the near future. Presumably, it will emphasize the design and qualification of an IHX capable of operating at 950 ^{o}C for long lifetimes (several decades). While some DDNs related to the IHX are generic (e.g., the materials qualification DDNs), other DDNs are design specific (e.g., printed circuit vs. helical coil, etc.); consequently, a reference conceptual design for the IHX is urgently needed to provide direction and priority to the energy transfer R&D programs. This Energy Transfer TDP also needs to address DDNs related process heat exchangers (hydrogen plants), piping insulation, isolation valves, and high temperature circulators.

1.4.4 Power Conversion System Technology Program

The NGNP program has not prepared a PCS TDP at this writing. Analogous to the Energy Transfer TDP, the scope of a PCS TDP will be strongly influenced by the PCS design. The NGNP PMPP and the ITRG have expressed a preference for an indirect-cycle PCS based upon conventional combustion-turbine technology with the implication that little technology development would be necessary. The GA team strongly recommends a direct-cycle PCS for the reasons elaborated in the PCDSR; in particular, an optimized direct-cycle plant will have significantly lower busbar cost. Specifically, the GA team recommends the direct-cycle PCS being designed and developed by OKBM under the DOE/ROSATOM International GT-MHR Program in Russia. OKBM, in collaboration with GA and ORNL, is conducting a comprehensive technology demonstration program to qualify this PCS design. GA believes that this PCS technology demonstration program will establish the technical viability of the design before the end of NGNP Preliminary Design.

1.4.5 Design Verification & Support Programs

The base technology for designing most MHR systems, structures and components derives from five decades of international R&D programs combined with the design, construction and operation of seven He-cooled reactors. Nevertheless, there are design-specific features of some SSCs that will require design verification by testing with semi-scale mockups or with actual prototypical components.

The current NGNP and NHI technology development programs are largely generic because there is no reference NGNP design. Many fundamental design selections have yet to be made:

reactor core type, IHX configuration, hydrogen production process, etc. Consequently, the current TDPs do not address DV&S DDNs to a significant degree. When the reference NGNP design is chosen, additional TDPs will need to be prepared that address the DV&S DDNs for key SSCs. It is expected that new design-specific TDPs will be needed for the Reactor System, Vessel System, Reactor Cavity Cooling Systems, etc.

Additional validation of the nuclear design methods will probably be needed for licensing purposes for the MHR design because of its annular core, which uses reflector control rods, and because of its reliance on inherent safety features, especially a strong negative temperature coefficient of reactivity, in contrast to engineered safeguards. The need to conduct new critical experiments, especially at elevated temperatures, will be problematic because no test facility currently exists in the USA. The only viable option would be to perform the tests in a foreign facility.

1.4.6 Hydrogen Production Programs

Nuclear hydrogen production technologies are being developed under the DOE Nuclear Hydrogen Initiative. The technology development programs, which have only been developed at a high level, are described in the Nuclear Hydrogen Initiative Ten Year Program Plan. The NHI plan covers both thermochemical water splitting and HTE (based on planar-cell technology). As presented in the NGNP PMPP, the NHI plan schedule is generally consistent with the NGNP construction schedule. The NHI Plan appears to address the SI and HTE DDNs; however, the plan lacks specificity. Both SI and HTE processes are considered to be immature for the reliable production of commercial quantities of hydrogen. Consequently, the planned construction and successful operation of small-scale and engineering-scale pilot plants for both SI and HTE will be critically important for the timely success of the NGNP hydrogen production mission. More detailed technology development plans for the SI and HTE processes should be developed during the Conceptual Design phase to ensure that the DDNs will be satisfied, especially those related to process integration and scale-up.

1.4.7 Design Methods Development and Validation

An extensive code development and validation program is presented in the NGNP Design Methods Development and Validation Research and Development Program Plan. The emphasis is heavily upon core nuclear and thermal/fluid dynamic computational methods. Design methods for predicting coated-particle fuel performance and radionuclide transport are not addressed; the Plan states that AGR fuel program will provide the necessary design methods for those applications. While the AGR Plan does include development of improved component models, etc., it does not include scope for developing advanced computational tools for full-core performance analysis or for predicting RN transport throughout the plant, and tritium transport is not addressed at all.

From GA team's perspective, the emphasis in this NGNP methods development plan is misguided. At least for prismatic-core MHRs, the currently available computational tools for core nuclear analysis and thermal/fluid flow analysis are largely adequate for NGNP Conceptual and Preliminary Designs. The traditional GA design methods for analyzing prismatic HTGRs, that were first developed to support the design and licensing of FSV and the large HTGRs in the 1970s, are still available. However, for nuclear analysis, the traditional codes have been largely supplanted by industry standard codes such as DIF3D and MCNP, and for thermal, flow, and structural analyses, commercial codes such as ANSYS, RELAP5, SINDA/FLUENT, and CFX, are already being used routinely by the GA team. In contrast, the design methods for predicting fuel performance and RN transport are in need of modernization and upgrading to support NGNP design and licensing.

1.5 Potential for International Collaboration

There is an impressive history of successful international collaboration on HTGR development, especially in the fuel, fission products and graphite areas. Arguably, the first major international cooperation on HTGR development – the Dragon Project – remains the most ambitious and successful one. One obvious and important difference in the on-going international modular HTGR programs is the choice of core design with the RF GT-MHR program and the Japanese program having selected a prismatic core and the PBMR and Chinese programs having chosen a pebble-bed core. While this difference is a complication in some regards, history is reassuring and encouraging. The US (prismatic core) and Germany (pebble-bed core) had a very productive cooperative program for gas-cooled reactor development, beginning in the late 1970s and continuing until the FRG HTR program was terminated in the late 1980s. Without exception, the greatest impediment to international collaboration is not technical differences but rather the establishment of government-to-government implementing agreements, especially regarding intellectual property rights.

The RF GT-MHR and the NGNP share many common DDNs, and much of the on-going RF technology program would be directly supportive of the NGNP Project. The OKBM Power Conversion Unit (PCU)[2] design is part of the GA NGNP preconceptual design. The OKBM design will have to be modified for 950 ^{o}C operation and to address the issues raised by the Rolls-Royce independent review. Much of the fuel, fission product and graphite technology programs should be directly relevant as well.

In addition to having common DDNs that the on-going RF technology programs could address for the NGNP Project, DOE/NNSA is providing half of the funding for the RF GT-MHR program. As a result, many of the intellectual property issues and QA pedigree issues that typically complicate international collaboration on nuclear construction projects should, in principle, be more tractable (e.g., OKBM is already ISO 9000 certified).

Japan has had an active interest in HTGR technology for decades. Presently, the Japan Atomic Energy Agency (JAEA) is conducting VHTR and nuclear hydrogen design and technology development. JAEA also operates the 30-MW(t) prismatic-core High Temperature Engineering Test Reactor. The HTTR could generate unique data to support the NGNP design and

[2] In this TDP, “PCU” indicates the OKBM design, and “PCS” indicates that design adopted for use in the NGNP, including those modifications needed for 950 ^{o}C operation.

licensing, especially regarding Ag and Cs release and plateout and an overall tritium mass balance for the plant. JAEA eventually plans to couple a SI-based hydrogen production plant to the HTTR using 10 MW of heat supplied from the HTTR IHX.

Nuclear Fuel Industries, which manufactured the HTTR fuel, has the only fuel manufacturing facilities in the world currently capable of mass producing TRISO LEU UO_2 fuel for the NGNP initial core in time for a 2018 startup.

In addition, Toshiba Corporation is developing the HTE process for hydrogen production. Toshiba and Fuji Electric, who designed the HTTR core, are both on the GA team, and a Toshiba design for an HTE plant is part of the GA “reference” NGNP preconceptual design. There is great potential for collaboration between the USA and Japan on H2-MHR design and development, but there is no government-to-government agreement for such collaboration in place at this writing.

In 2004, the Republic of Korea initiated a project to develop hydrogen production using the VHTR and the SI process. VHTR design and technology development is being performed by the Korea Atomic Energy Research Institute (KAERI) and development of the SI process is being performed by the Korea Institute of Energy Research (KIER) and the Korea Institute of Science and Technology (KIST). DOOSAN Heavy Industries & Construction is also participating in the project, which is known as the Nuclear Hydrogen Production and Technology Development and Demonstration Project (NHDD).

In August 2005, a Memorandum of Understanding was signed between GA and KAERI/DOOSAN, which included establishing Nuclear Hydrogen Joint Development Centers (NHJDC) in both San Diego and Daejeon, Korea. Current areas of collaboration include SI process development and modelling, VHTR core design and optimization, vessel cooling, fuel performance and fission product transport, tritium source terms and impacts on hydrogen production, fuel manufacturing, availability/reliability, seismic analyses, availability/reliability assessments, and investigation of deep-burn fuel cycles.

The European Union sponsors a number of projects to promote MHR technology, the most significant of which is RAPHAEL. The RAPHAEL project will investigate the performance of fuel, materials and components, the reactor physics models, the nuclear safety and waste disposal issues, and the overall system integration. In addition to these base programs for HTR development, AREVA and CEA are conducting R&D programs in support of the ANTARES VHTR design (prismatic core). Much of this European workscope would be directly relevant to the NGNP Project. However, there is at present no non-European participation in these programs. The potential for collaboration between NGNP and RAPHAEL is unknown at this time.

Of the European test facilities, the High Flux Reactor (HFR) Petten, NL, is of particular interest. This reactor was used extensively by the former German TRISO fuel development program. Consequently, they have fully qualified multi-capsule test rigs available; however, their on-site capability for performing postirradiation examinations of coated-particle fuel is rather limited. A

US in-pile fission product transport test was also irradiated in HFR Petten. Presumably, HFR Petten would be available to support the NGNP program on a contract basis.

The PBMR Project has planned and is conducting a significant R&D program to support the design and licensing of their prototype pebble-bed module. Many DDNs, especially those relating to fuel, fission products, graphite, and high-temperature metals, are generic. Technically, there is great potential for collaboration between PBMR and NGNP. The impediments to collaboration with PBMR appear to be commercial (e.g., intellectual property rights) and political rather than technical. Once the NGNP conceptual design has been chosen, the prospects for collaboration should be revisited.

1.6 Cost and Schedule

Published cost estimates for the various technology programs from FY06 through completion total $1,029,130, and certain cost elements are obviously missing (e.g., qualification of NFI UO_2 fuel for the initial core). Taken at face value, the existing cost estimates imply that the NGNP technology development programs need to be reprioritized. Arguably, the highest priority technology task is the qualification of UCO fuel and the establishment of the technical basis for the design and construction of a domestic fuel fabrication facility. However, only 13% of the total cost estimate is for UCO fuel qualification. Even more striking, only 3% of the total is for validating the radionuclide source terms which will be essential for licensing.

In general, it is not possible to critique the existing schedules and cost estimates with any confidence because the corresponding workscopes are not defined in sufficient detail to permit an independent estimate. This circumstance is especially problematic with the NHI 10-yr R&D Plan for SI and HTE development.

The exception is the AGR fuel development plan wherein the workscope is generally well defined. However, there are two significant problems with the AGR plan. First, the development schedule is not supportive of the NGNP design and licensing schedule required for a 2018 startup. For example, the Final Design phase would need to be completed by the end of FY13, but the safety testing and the source term validation tasks are not scheduled for completion until FY19. Secondly, the AGR plan is missing workscope which would substantially increase the total program costs: (1) qualification of NFI UO_2 fuel for the initial core and early reloads; (2) an integrated fuel pilot plant to provide the technical basis for a NGNP fuel fabrication facility; and (3) a test program to characterize RN transport in the VLPC.

The PMPP estimates that the cost for NFI process development and fabrication of AGR-2a test fuel would be ~$ 6M, and that the cost for irradiation, safety testing and PIE for AGR-2a would be ~$11M. The cost for an integrated fuel fabrication pilot plant would be considerably more and would depend upon the design throughput (e.g., the number of coaters, etc.). As an indication, a fuel fabrication facility with a throughput of 510 fuel elements/year (i.e., a reload segment for a 600 MW(t) NGNP per year) has been estimated to cost ~$200M.

The task of focusing and prioritizing the technology programs will become more straightforward once the NGNP design is officially determined. At that time, a stand-alone, bottoms-up umbrella

TDP needs to be prepared which is responsive to the NGNP DDNs and to the design and licensing schedule. Presumably, the total R&D costs can also be reduced significantly.

Table 1-1. Key NGNP DDNs and Technology Development Requirements

Technology Area	Key DDNs	Applicable TDP	TDP Responsive To DDNs?	Major Facility Deficiencies	Recommended Resolution
UCO Fuel Qualification	– Fabrication process integration and scale-up – Qualification of NFI 10%-enriched UO2 for initial core – UCO irradiation performance – UCO performance during core heatup	AGR Fuel Plan	– Schedule does not support 2018 NGNP start-up – No integrated fuel fabrication pilot plant – No qualification of NFI UO_2 fuel for initial core	– Limited irradiation capacity in ATR – Limited Hot Cell capacity at ATR – No in-pile facility for reactivation of irradiated fuel – No in-pile facility for R/B measurements	– Increase priority of AGR irradiations in ATR – Consider contracting with HFR Petten – Upgrade Hot Cells – Install King Furnace in NRR TRIGA – Add fuel pilot plant to AGR program – Add NFI fuel to AGR program
Radionuclide Transport	– Ag & Cs release from core – Ag & Cs plateout on turbine and IHX – I-131 release during core heatup accidents – Integral test data for methods validation	AGR Fuel Plan	– Schedule does not support NGNP 2018 startup – Plan for in-pile RN transport loop unrealistic – No RN transport in VLPC	– No in-pile RN transport loop – No facility for RN transport in VLPC tests	– Increase priority of RN transport tasks – Evaluate feasibility of loop in ATR – Consider contracting with NIIAR, RF, for use of PG-1 loop – Add VLPC mockup to in-pile loop
Spent Fuel Disposition	– “Non-combustibility” of graphite – Long-term leaching of irradiated TRISO fuel – C-14 production and transport	LEU Spent Fuel TDP	– Spent Fuel TDP not in PPMP	– None identified	– Lab review of Spent Fuel TDP – Add Spent Fuel TDP to PPMP

Technology Area	Key DDNs	Applicable TDP	TDP Responsive To DDNs?	Major Facility Deficiencies	Recommended Resolution
Core Graphite	– Initial screening of candidate graphites – Qualification of replacement for H-451	NGNP Materials R&D TDP	– TDP appears responsive to DDNs – Excessive number of candidate graphites	– Limited capacity for graphite irradiation	– Prepare stand-alone graphite TDP – Use capsule AGC-1 as screening capsule to reduce number of graphites – Focus on qualifying H-451 replacement graphite – Irradiate graphites at NIIAR as necessary
High-Temperature Metals	– IN 617 for IHX @ 950 C – 2¼Cr-1Mo for RPV – 9%Cr-1Mo-V as backup to 2¼Cr-1Mo for RPV – Turbine blade alloys for 950 °C (e.g., IN 100)	NGNP Materials R&D Program	– All metals DDNs addressed, except… – Turbine blade alloys need to be added – Testing needs to be prioritized (fewer candidates)	– None identified	– Add turbine blade alloys – Prioritize tasks and focus on prime and backup alloy for each application
Power Conversion System	– 950 °C operation without blade cooling – Recuperator lifetime – Large EM bearings	RF GT-MHR PCU TDP	– All DDNs addressed, except… – Additions needed for 950 °C operation – Full-scale PCS prototype to verify design?	– None identified	– Address Rolls-Royce critique of OKBM design – Add scope for 950 °C operation – Monitor OKBM PCU technology program
Design Verification & Support (DV&S)	– Nuclear criticals (annular core) – Integrated RCCS performance – High temperature circulators	No DV&S TDPs	None	– No suitable critical facility in USA – No large, high temperature He loop for testing of prototypical components	– Prepare DV&S TDPs – Use RF ASTRA critical facility – Use OKBM He loops (3)
H_2 Production	– Reaction kinetics	NHI 10-yr R&D	– SI DDNs apparently	– None identified	– Prepare quantitative

Technology Area	Key DDNs	Applicable TDP	TDP Responsive To DDNs?	Major Facility Deficiencies	Recommended Resolution
– SI Process	— Process integration & scale-up — Structural materials corrosion	Plan	addressed; plan too qualitative for certainty – Pilot-plant testing critically important		SI TDP — Collaborate with JAEA and KAERI on SI process development — Expedite pilot-plant testing
H_2 Production – HTE Process	— SOEC sealing — Production costs — SOEC service lifetime — Spent SOEC disposition	NHI 10-yr R&D Plan	— HTE DDNs apparently addressed; plan too qualitative for certainty — Pilot-plant testing critically important	— None identified	— Prepare quantitative HTE TDP — Collaborate with Toshiba on HTE process development — Expedite pilot-plant testing
Design Methods Validation	— Single-effects tests for improved models/properties — Independent integral tests for code validation	NGNP Methods Development Plan	— [Normally, design workscope; not technology task] — Overemphasis on nuclear and thermal/flow codes — Fuel performance & RN transport not included	— No suitable critical facility in USA — No high temperature He loop — No in-pile RN transport loop	— Use industry standard codes to the extent applicable — Prepare V&V Plan for RN control codes — Use RF test facilities as appropriate — Use reactor surveillance data from HTTR to the extent available

2. INTRODUCTION AND BACKGROUND

The US Department of Energy (DOE) has chosen the Very High Temperature Reactor (VHTR) for the Next Generation Nuclear Plant (NGNP) Project. The reactor design will be a helium-cooled, graphite-moderated thermal reactor that will be designed to produce electricity and hydrogen as required by the Energy Policy Act of 2005 (EPACT). DOE has contracted with three industrial teams, including a team led by General Atomics (GA), for preconceptual design engineering services (Work Plan 2006). As part of the contractual work scope, GA has prepared this umbrella technology development plan (TDP) which documents the research and development (R&D) that must be performed on a timely schedule to support NGNP design and licensing. To accomplish this task (identified as WBS Element 1600 in the Work Plan), GA defined the Design Data Needs (DDNs) to assure the NGNP design meets the requirements defined in the System Requirements Manual (SRM 2007) and evaluated the extent to which the DOE-sponsored R&D programs being performed by the NGNP Project and the Nuclear Hydrogen Initiative (NHI) will satisfy these DDNs.

2.1 Purpose and Scope

As described in the Preliminary Project Management Plan (PPMP 2006), a key goal of the NGNP Preconceptual Engineering Services contracts is to focus and prioritize research and development activities for the NGNP to maximize their support of plant design and licensing. The primary purpose of this TDP is to provide that focus and prioritization for those R&D programs needed to support the NGNP Project.

The first step was to specify an NGNP “reference” preconceptual design, which is documented in the Preconceptual Design Studies Report (PCDSR 2007), because many DDNs are design-specific (other DDNs such as those related to fuel and fission products are largely generic). The status of the various technologies needed to support design and licensing was reviewed, and DDNs were defined where the current database was judged to be inadequate. Many of the resulting NGNP DDNs, especially those related to the Reactor System (RS) and Power Conversion System (PCS), are very similar to the DDNs for the commercial Gas Turbine-Modular Helium Reactor (GT-MHR), but new DDNs related to the intermediate heat exchanger and hydrogen production were also identified.

The DOE-sponsored technology programs intended to support the NGNP, including the various NGNP R&D programs and the Nuclear Hydrogen Initiative (NHI) programs, were then evaluated, and their responsiveness to the DDNs was assessed. The existing TDPs were critiqued on an exception basis to identify any deficiencies and unnecessary workscope, especially in the context of the NGNP schedule. In general, these TDPs propose to investigate an excessively large number of materials (graphite, metals, etc.) and so the candidate materials need to be prioritized.

Other international HTGR development programs were also reviewed to determine the potential benefit of international collaboration to the NGNP Project. In particular, the DOE/ROSATOM-sponsored, International GT-MHR program for weapons Pu (WPu) disposition shares many common DDNs with the NGNP, especially with regard to the Reactor System and Power Conversion System. The Japanese HTGR development program, especially the High Temperature Engineering Test Reactor (HTTR) surveillance programs, is also highly relevant.

The results of these assessments are documented in this TDP. In large measure, it is a critique of the existing TDPs, consistent with the purpose of providing focus and prioritization to the NGNP R&D programs, rather than a stand-alone, bottoms-up TDP. While this current TDP is considered appropriate for preconceptual design, an integrated umbrella NGNP TDP should be prepared during the Conceptual Design phase and periodically updated as the NGNP design matures and feedback is obtained from licensing authorities and potential customers.

2.2 Programmatic Overview

The following subsections are intended to provide a context in which the technology development needs for NGNP design and licensing can be assessed. Much of the information is provided by reference, especially to the PCDSR.[3] The required plant design and licensing schedule is based upon planning Option 2 (“Balanced Risk”) in the NGNP PPMP which requires a 2018 plant startup.

2.2.1 NGNP Project Goals

As defined in the NGNP PPMP (2006), the NGNP Project objectives that support the NGNP mission and DOE’s vision are as follows:

a. Develop and implement the technologies important to achieving the functional performance and design requirements determined through close collaboration with commercial industry end-users

b. Demonstrate the basis for commercialization of the nuclear system, the hydrogen production facility, and the power conversion concept. An essential part of the prototype operations will be demonstrating that the requisite reliability and capacity factor can be achieved over an extended period of operation.

c. Establish the basis for licensing the commercial version of the NGNP by the Nuclear Regulatory Commission (NRC). This will be achieved in major part through licensing of the prototype by NRC, and by initiating the process for certification of the nuclear system design

d. Foster rebuilding of the US nuclear industrial infrastructure and contributing to making the US industry self-sufficient for its nuclear energy production needs

Additional objectives that are not explicitly stated in the PPMP, but that should be considered applicable to the NGNP include:

e. Provide a level of safety assurance that meets or exceeds that afforded to the public by modern commercial nuclear power plants

[3] In addition to being a stand-alone document, this TDP is also included as an attachment to the PCDSR and is summarized in Section 7.3 of that document.

f. Meet or exceed all applicable federal, state, and local regulations or standards for environmental compliance

GA is in agreement with the above objectives but believes that the NGNP Project as described in the PPMP is missing a program element that is critical to achieving objectives b and d. Specifically, GA believes that the NGNP Project should include a demonstration of the viability of commercial-scale, coated-particle fuel fabrication. GA also believes that the NGNP mission should be expanded to include demonstration of the MHR "Deep Burn" concept that has been proposed by GA for destruction of Russian weapons-grade Pu and the transuranic waste from LWRs. GA's view is that there is much to be gained by coordinating the NGNP Project with the Russian Pu-disposition Program and with the Global Nuclear Energy Partnership (GNEP), but it appears that the potential for the NGNP to play an important role as a test bed for demonstrating the irradiation performance of Deep Burn fuel is being largely overlooked by DOE

2.2.2 NGNP Preconceptual Design

The GA "reference" preconceptual design for the NGNP is presented in detail in the PCDSR (2007). The essential features are illustrated in Figures 2-1 through 2-8. Figure 2-1 shows a simplified schematic of the NGNP preconceptual plant design, and Table 2-1 summarizes the key design features of the plant.

2.2.2.1 Nuclear Heat Source

The nuclear heat source for the NGNP consists of a single 600-MW prismatic-block MHR module with two primary coolant loops for transport of the high-temperature helium exiting the reactor core to a direct cycle power conversion system and to an intermediate heat exchanger (IHX). As shown in Figures 2-2 and 2-3, the reactor design is essentially the same as for the GT-MHR (Shenoy 1996) but includes the additional primary coolant loop (Figure 2-4) to transport heat to the IHX and other modifications to allow operation with a coolant-outlet temperature of 950°C (vs. 850°C for the GT-MHR). The IHX transfers 65 MW of thermal energy to the secondary heat transport loop, which transports the heat energy to both a SI-based hydrogen production facility (60 MW) and an HTE-based hydrogen production facility (4 MW). The basic components of a prismatic fuel element are shown in Figure 2-5.

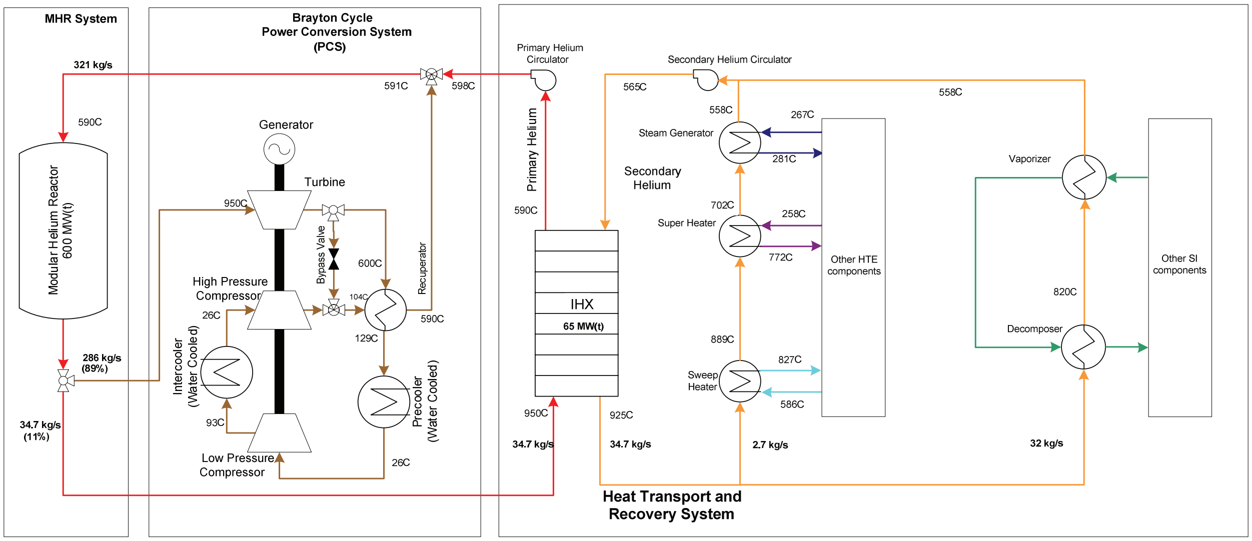


Figure 2-1. Schematic of NGNP Plant

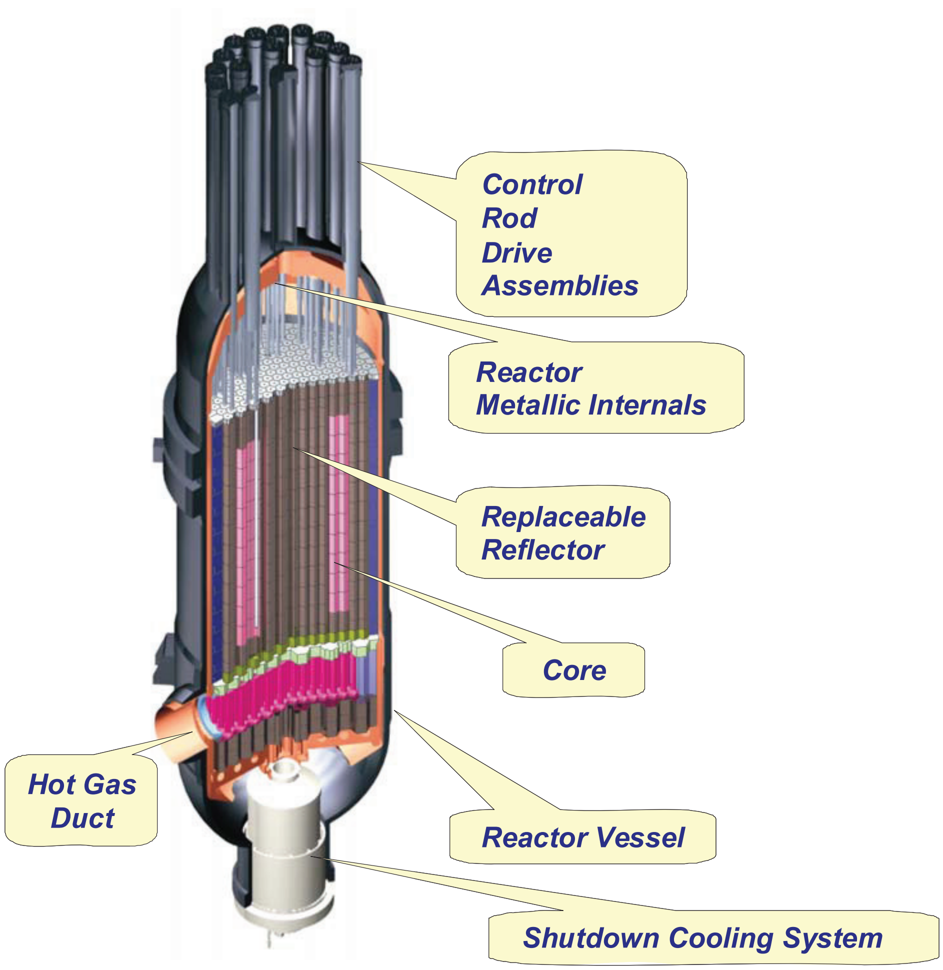


Figure 2-2. MHR Nuclear Heat Source

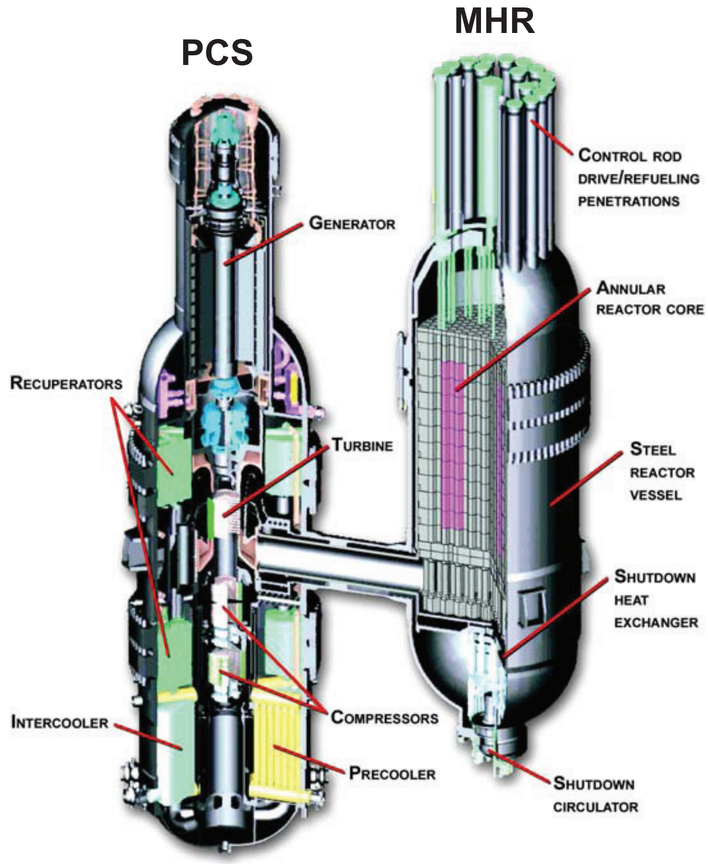


Figure 2-3. GT-MHR for Electricity Production

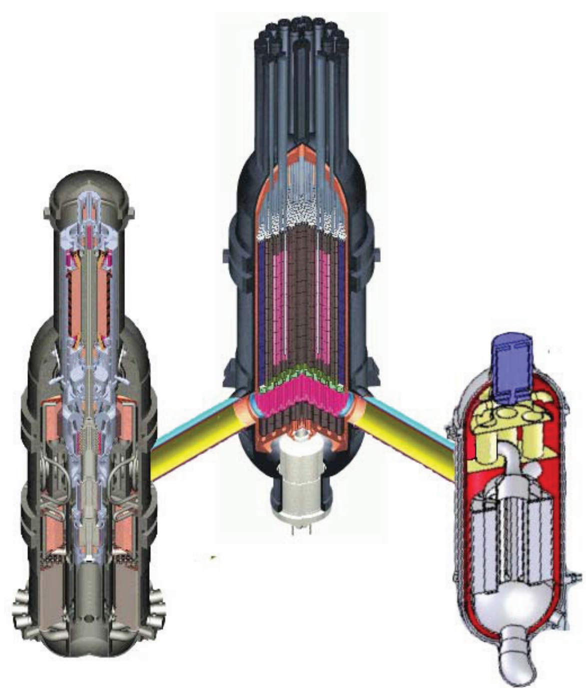


Figure 2-4. MHR Module Connected to a Direct-Cycle PCS and an IHX

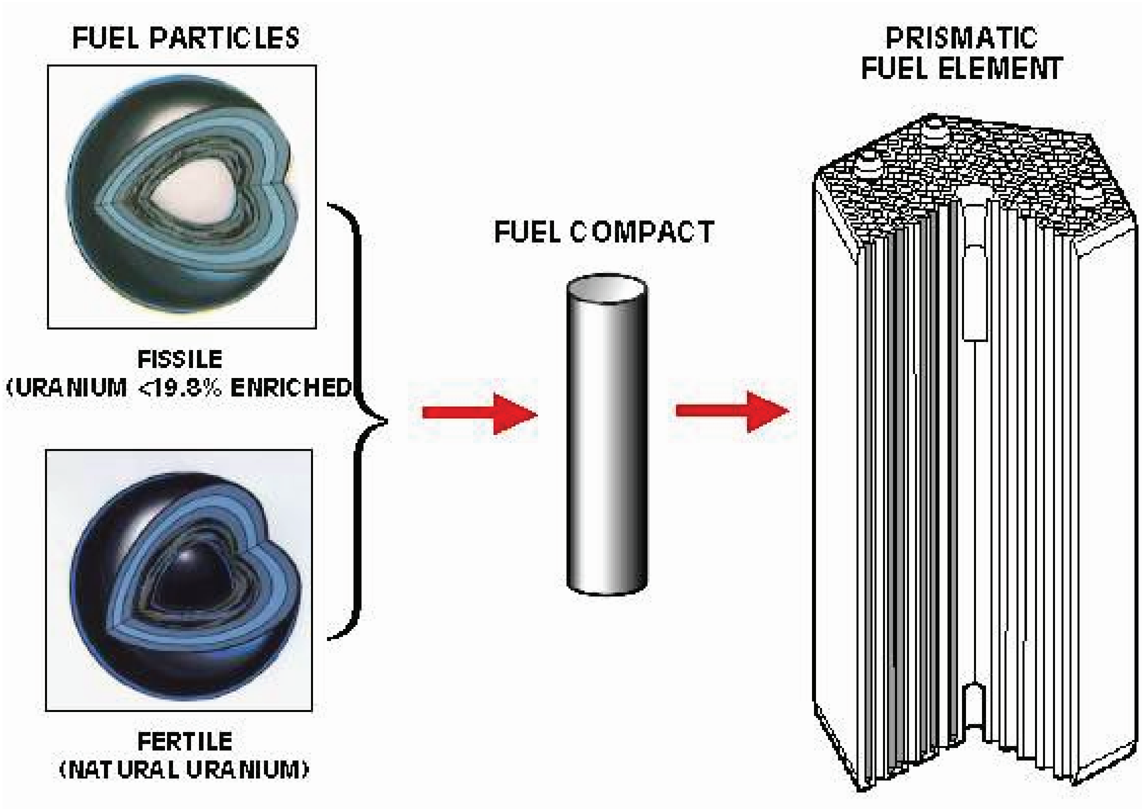


Figure 2-5. Components of a Prismatic Fuel Element

2.2.2.2 Hydrogen Production Processes

Hydrogen will be produced in the NGNP by both sulfur-iodine (SI)-based thermochemical water splitting and by high temperature electrolysis (HTE). A modular design will be used for both plants such that the modules can be replicated to provide a commercial-sized plant based upon either technology.

The essential enabling features of the two hydrogen production technologies are summarized below. The NGNP hydrogen plant designs are described in the PCDSR. The status of these two technologies is discussed in Section 3.

2.2.2.2.1 SI-based Thermochemical Water Splitting

Water thermally dissociates at significant rates into elemental hydrogen and oxygen only at temperatures approaching 4000 °C. As indicated in Figure 2-6, the SI process consists of three primary chemical reactions that accomplish the same result at much lower temperatures.

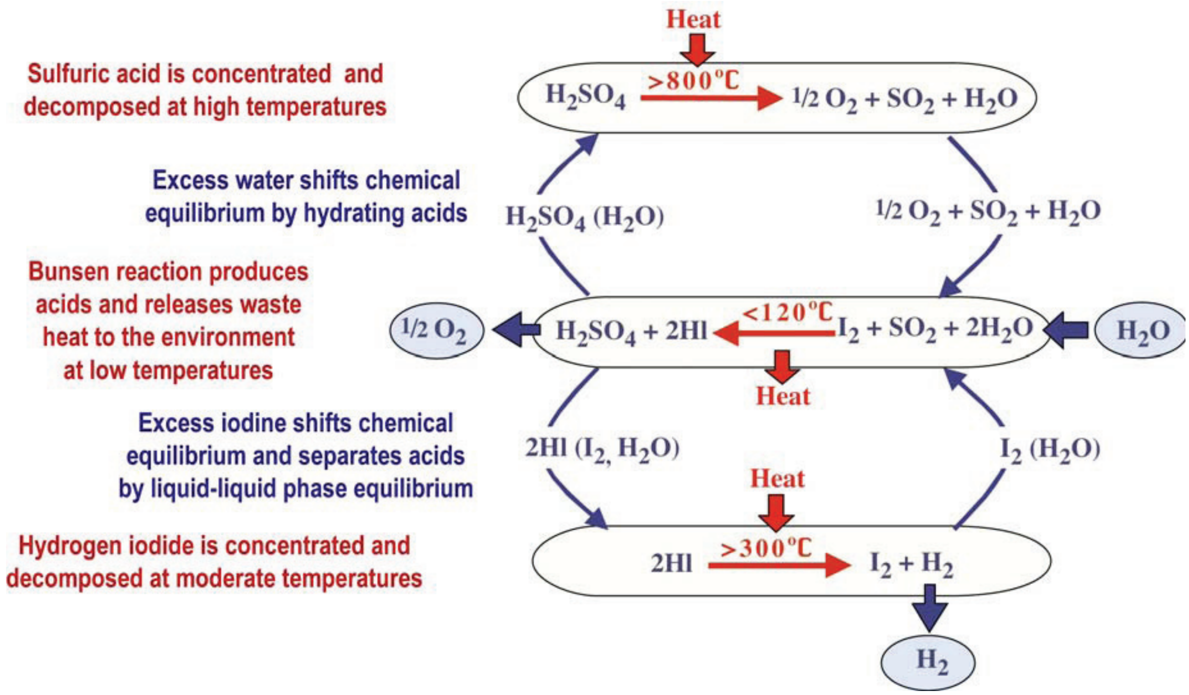


Figure 2-6. The SI Thermochemical Water Splitting Process

The process involves decomposition of sulfuric acid and hydrogen iodide and regeneration of these reagents using the Bunsen reaction. Process heat is supplied at temperatures greater than 800 °C to concentrate and decompose sulfuric acid. The exothermic Bunsen reaction is performed at temperatures below 120 °C and releases waste heat to the environment. Hydrogen is generated during the decomposition of hydrogen iodide using process heat at temperatures greater than 300 °C. Two different processes are being investigated for HI decomposition. One process, referred to as reactive distillation, involves reacting the HI-water-iodine mixture in a reactive bed to effect the separation process and produce hydrogen. The other process, referred to as extractive distillation, uses phosphoric acid to strip HI from the HI-water-iodine mixture and to break the HI-water azeotrope. The NGNP preconceptual design is based upon the latter process since the reaction kinetics for the former appear unfavorable.

2.2.2.2.2 High Temperature Electrolysis

Electrolysis is performed at high temperatures using solid oxide electrolyzer (SOE) modules. Figure 2-7 shows a schematic diagram of a unit solid oxide electrolyzer cell (SOEC). Conceptually, a solid oxide electrolyzer cell is a solid oxide fuel cell (SOFC) operating in reverse.

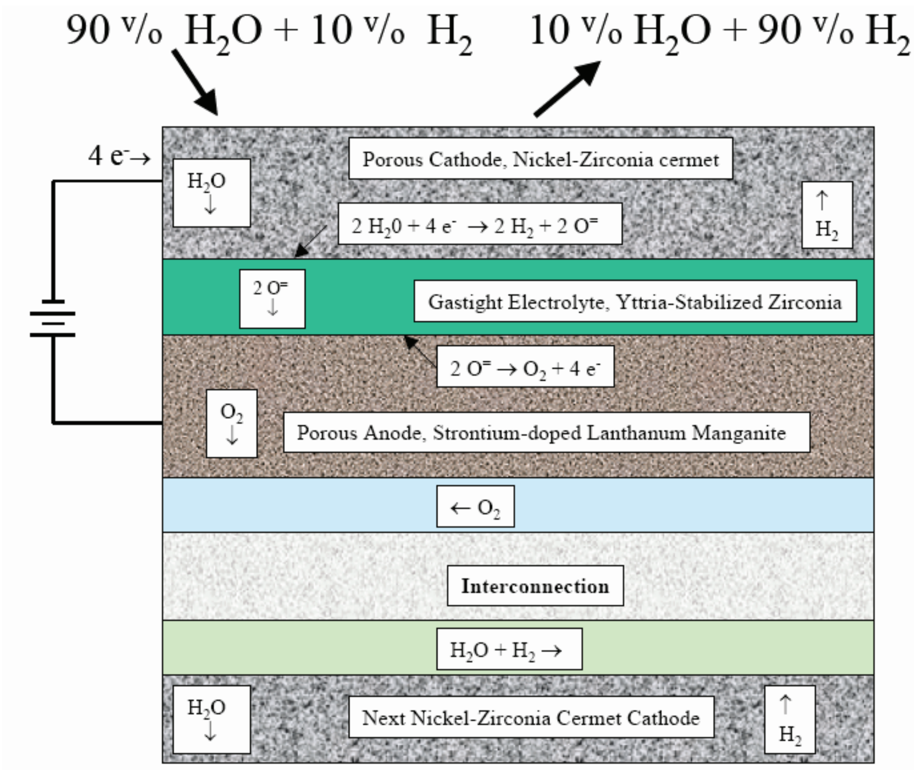
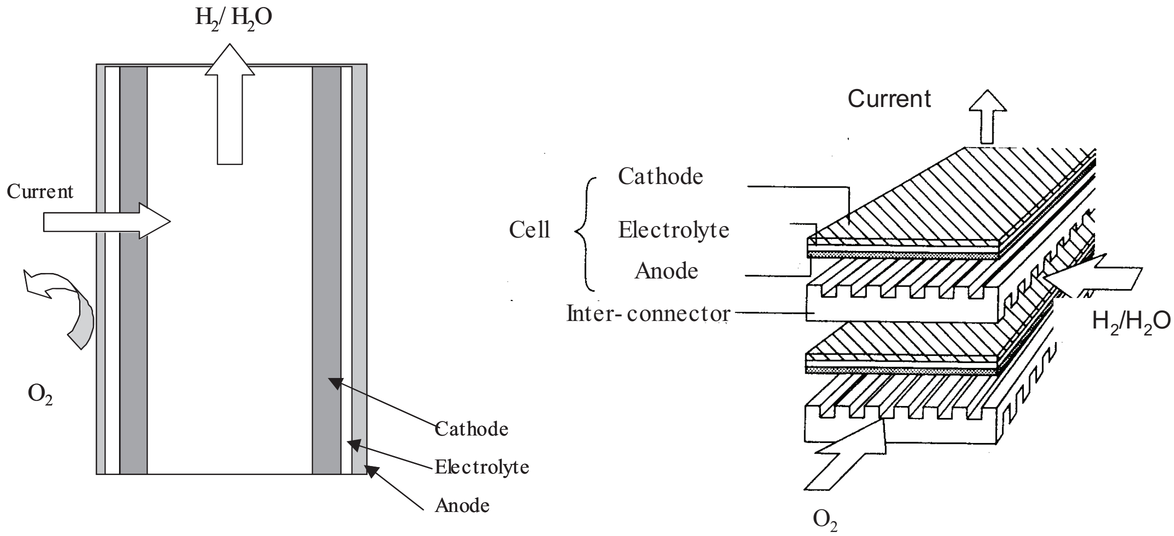


Figure 2-7. Solid Oxide Electrolyzer Cell

Steam is supplied to both the anode and cathodes sides of the solid oxide electrolyzers. The steam supplied to the cathode side is electrolytically split into hydrogen and oxygen. The oxygen is transferred through the electrolyte to the anode side. The steam supplied to the anode side is used to sweep the oxygen from electrolyzer modules. The steam supplied to the cathode side is first mixed with a small portion of the hydrogen stream to ensure reducing conditions and prevent oxidation of the electrodes.

The cell electrolyte is fabricated from either yttria- or scandia-stabilized zirconia. A number of materials are being investigated for use as the anode and cathode. In one leading design, a 1.5-mm cathode plate made of nickel cermet material is bonded to one side of the electrolyte. A 0.05 mm anode plate is bonded to the other side of the electrolyte. The anode is composed of a mixed (i.e., both electronic and ionic) conducting perovskite, lanthanum manganate ($LaMnO_3$) material. Bipolar plates with a doped lanthanum chromite (e.g., $La_{0.8}Ca_{0.2}CrO_3$) are attached to the outside of the anode and cathode, and join the anode and cathode of adjacent units to form a stack.

Two different cell configurations are under active development: (1) a planar-cell technology being developed as part of a collaborative project between INL and Ceramatec of Salt Lake City, UT, and (2) a tubular-cell technology being by Toshiba Corporation. The two configurations are compared in Figure 2-8.



Tubular type   Planar type

Figure 2-8. Comparison of Tubular-Type and Planar-Type HTE SOECs

Each configuration has its advantages and disadvantages. For example, a potential issue for the planar-cell concept is stack durability and compromised sealing as the result of thermal cycling. Tubular cells have less active cell area per unit volume than planar cells but are less susceptible to this issue. The GA team believes both the planar-cell and tubular-cell technologies are promising concepts for future commercialization and recommends that both concepts be developed through at least the pilot-scale demonstration stage.

Table 2-1. Key Features of NGNP Preconceptual Design

Property	Design Selection
Reactor type	Prismatic block
Reactor power level	600 MW(t)
Fuel	Initial Core: TRISO-coated 500-μm UO_2 (~9.9% enriched) Reloads: TRISO-coated UCO (with two or more different U-235 enrichments or a natural UCO fertile particle)
Power conversion cycle	Reference: Direct Brayton cycle with GA/OKBM vertical integrated PCS Alternate: Direct combined cycle (steam cycle with a GT topping cycle) as proposed by Rolls-Royce
Core outlet/inlet coolant temperatures	Reference: 950ºC/590ºC Alternate: 950ºC/510ºC (to be evaluated as a means of keeping RPV temperatures low enough to use SA508/SA533 steel)
Pressure vessel materials	RPV: 2¼Cr-1Mo PCS: SA508/SA533 IHX: 2¼Cr-1Mo Cross duct outer pipe: 2¼Cr – 1Mo Cross duct inner pipe: Allow 617 or alloy 800H
Primary loop inlet/outlet pressure	7.07MPa/7.0 MPa
Number of loops	3 (PCS loop, primary heat transport loop, and secondary heat transport loop)
Primary coolant	Helium
Secondary loop working fluid	Helium
Heat transferred to secondary loop	65 MW(t)
Intermediate heat exchanger type (and LMTD)	Reference: Printed circuit (25°C) Backup: Helical coil (TBD)
Reactor Cavity Cooling System	Air-cooled RCCS
Reactor building	Vented Low-Pressure Containment (VLPC)
Hydrogen production process	SI requiring 60 MW(t) thermal energy HTE requiring 4 MW(t) thermal energy
Heat rejection	Dry cooling towers

2.2.3 Relationship between Design and Technology Development

As described in the NERI reports (Richards 2006a and Richards 2006b), GA uses the protocol illustrated in Figure 2-9 for integration of design with technology development in order to maximize the benefit of the technology development programs in terms of supporting a plant design and minimizing the technical risk of the design. This model is based on successful Engineering Development and Demonstration (ED&D) programs conducted and managed by GA for DOE projects, including Accelerator Production of Tritium, the Salt Waste Processing Facility, the commercial GT-MHR, and the New Production Reactor.

As shown in Figure 2-9, the process begins by evaluating design requirements and reviewing existing design data from a variety of sources. Design assessments and trade studies are performed, eventually leading to key design selections and a technical baseline that meets all design requirements. As indicated on Figure 2-9, it may be reasonable to revise one or more design requirements during the process if the overall impact is small. At this point, a design has been developed that meets all requirements, but requires some technology development to confirm assumptions upon which the design is based. Also, if necessary, the process allows for a testing path to provide early confirmation of basic assumptions.

The technology development process begins with the design organization preparing DDNs, which are formal project documents that include fallback positions in the event the testing programs do not produce acceptable results or the test could not be performed for budgetary or other reasons. The DDNs provide a concise statement of the required data and the associated schedule, quality, and accuracy requirements. In addition to preparing DDNs, the design organization also prepares a test specification that defines the data requirements in more detail. The technology organization is responsible for developing Technology Development Plans and Test Plans for specific tests. As indicated on Figure 2-9, the design and technology organizations work together during preparation of the DDNs, test specifications, technology development plans, and specific test plans.

The technology organization then conducts the technology development programs and generates the design data. If feasible, the technology organization may integrate their activities with other (e.g., international) programs in order to minimize costs. After the design data are obtained, the design and technology organizations work together to determine if the DDNs are satisfied. If the DDNs are satisfied, the key design selections and technical baseline are finalized and the design is completed. If a DDN is not satisfied, the most likely path forward is to adopt the fallback position, which could mean additional margin is added to a certain area of plant design in order to reduce technical risk. However, depending on the results of a specific test program, a more reasonable path forward may be to re-evaluate a key design selection and return to the design process. As indicated on Figure 2-9, an independent review and verification organization is established at the start of the process to provide oversight of both the design and technology development processes

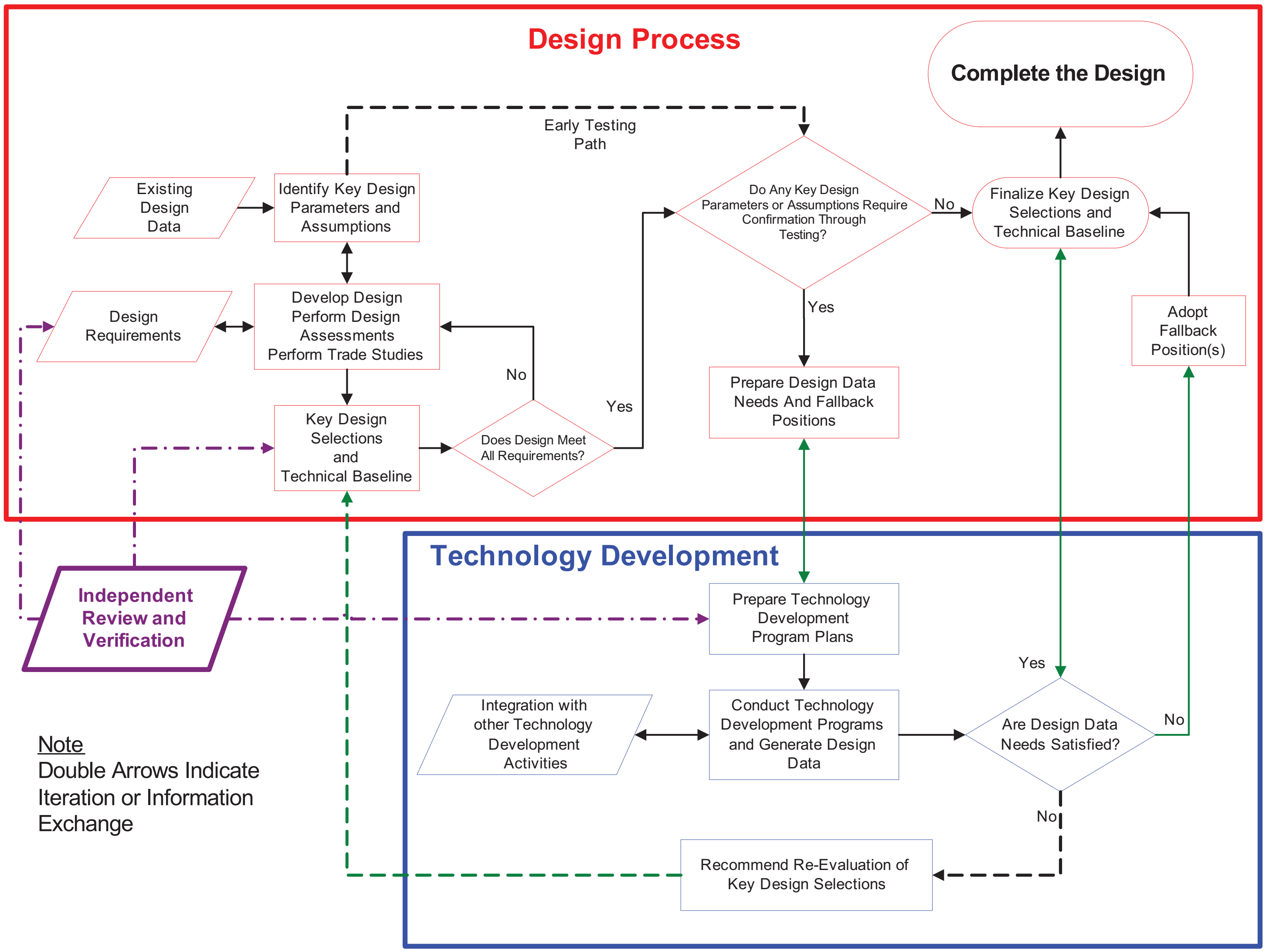


Figure 2-9. Integration of Design with Technology Development

2.2.4 Key NGNP Technology Development Issues

The Energy Policy Act of 2005 outlines five specific areas of research, called "Major Project Elements," that would support the NGNP project (EPACT Section 643 (a)(1-5)):

1. High-temperature hydrogen production technology development and validation
2. Power conversion technology development and validation
3. Nuclear fuel development, characterization, and qualification
4. Materials selection, development, testing, and qualification
5. Reactor and balance-of-plant design, engineering, safety analysis, and qualification.

The workscopes are largely intuitive from the titles and will be described in some detail in later sections of this TDP. The NGNP PMPP (2006) endorses these five research areas and adds a sixth one:

6. Energy transfer which includes the intermediate heat exchanger and the secondary heat transfer loop.

In principle, the GA team agrees that these are the priority R&D areas for the NGNP Project (with the caveat that no particular priority should be implied from the order in which they are listed). However, within each broad category, a detailed definition of the planned technology development is necessary before an assessment of its responsiveness to NGNP DDNs can be made. When such an assessment was made, a number of deficiencies were identified that are described in subsequent sections (e.g., turbine blade alloys should be included in the NGNP materials R&D program).

2.2.5 Potential for International Collaboration

There is currently considerable international interest in Modular Helium Reactors to contribute to the resolution of a broad spectrum of national and international issues. These other MHR programs are also conducting R&D programs that will generate data that, in principle, could be used to satisfy NGNP DDNs or portions thereof; hence, the potential for international collaboration is substantial. The international MHR programs that appear to have the greatest potential to support the NGNP Project are briefly described below.

The International GT-MHR is being developed under a joint USDOE-NNSA/ROSATOM program for the purpose of destroying surplus Russian weapons plutonium (CDR 1997). The reference plant design is very similar to the GA commercial GT-MHR design (Shenoy 1996) with an improved Power Conversion Unit (PCU) design. A preliminary design has been completed. The reference fuel particle design is TRISO-coated PuO_{2-x}, and construction of a bench-scale facility (BSF) for fabrication of Pu test fuel is nearly complete. The lead design organization OKBM is a member of the GA team, and the OKBM PCU is part of the GA "reference" preconceptual design.[4] The goal is to have a demonstration plant, consisting of two 600 MW(t)

[4] In this TDP, "PCU" refers specifically to the OKBM 850 ^{o}C design; "PCS" refers to the adaptation of the OKBM design for use in the NGNP.

modules in operation by 2019. The International GT-MHR and the NGNP share many common DDNs, and much of the Russian technology program (TDPP 2005) would be supportive of the NGNP Project.

The Japan Atomic Energy Agency (JAEA) is conducting an ambitious MHR development program, the cornerstone of which is the continuing operation of the HTTR. JAEA is currently performing laboratory-scale investigations of the SI hydrogen process with plans to construct a pilot plant in the near future and to couple an engineering-scale SI pilot plant with the HTTR in the 2011 timeframe. JAEA along with industrial partners has also produced preconceptual designs for a direct-cycle gas-turbine MHR operating at 850 ^{o}C and a VHTR operating at 950 ^{o}C that would produce electricity with a direct-cycle gas turbine and hydrogen via the SI process. Toshiba Corporation is developing the HTE process for hydrogen production. Toshiba and Fuji Electric are both on the GA team, and a Toshiba design for an HTE plant is part of the GA "reference" NGNP preconceptual design. There is great potential for collaboration between the USA and Japan on H2-MHR design and development, but there is no government-to-government agreement for such collaboration in place at this writing.

The Korea Atomic Energy Research Institute (KAERI), DOOSAN Heavy Industries and Construction, Ltd., and General Atomics have agreed to cooperate on the development of nuclear hydrogen production technologies using the VHTR. The Republic of Korea (ROK) intends to develop and demonstrate SI-based hydrogen process on an engineering scale under the Nuclear Hydrogen Development and Demonstration (NHDD) project. KAERI and their industry partners are performing plant design trade studies, developing reactor design methods, and are operating a laboratory-scale SI test facility. KAERI has made a detailed comparison of the prismatic and pebble-bed core designs (KAERI 2007), but an official selection of reactor type has not been made at this writing. KAERI is also a member of the GA team.

The South Africa-based Pebble Bed Modular Reactor (PBMR) project (e.g., Nicholls 2001) to construct a prototype of a commercial pebble-bed modular HTGR is also progressing. The PBMR is a 400 MW(t) modular, direct-cycle, pebble-bed MHR (Slabber 2005), which is based upon the pebble-bed reactor technology successfully developed and demonstrated in the Federal Republic of Germany (FRG). Under development since 1993, the PBMR project entails the building of a demonstration reactor module at Koeberg near Cape Town and a pilot fuel plant at Pelindaba near Pretoria. The current schedule is to start construction in 2009 and for the first fuel to be loaded four years later. Construction of the first commercial PBMR modules are planned to start three years after the first fuel has been loaded into the demonstration reactor. The GA "reference" NGNP preconceptual design and the PBMR share many common DDNs, related to fuel, fission products, graphite, high-temperature materials, etc. There is great potential for collaboration, but the political will for such has evidently been lacking to date.

2.3 Background

Brief descriptions of the MHR design philosophy and previous MHR designs and their associated technology development programs are provided below in order to provide a context for evaluating the technology development needs for the NGNP. (Most of the information is provided by reference.)

2.3.1 MHR Design Philosophy

A fundamental requirement for the design of any nuclear power plant is the containment and control of the radionuclides produced by various nuclear reactions; in response, different radionuclide containment systems have been designed and employed for different reactor designs. For modular HTGR designs, a hallmark philosophy has been adopted since the early 1980s to design the plant such that the radionuclides would be retained in the core during normal operation and postulated accidents. The key to achieving this safety goal is the reliance upon ceramic-coated fuel particles for primary fission product containment at their source, along with passive cooling to assure that the integrity of the coated particles is maintained even if the normal cooling systems were permanently disrupted.

This innovative design philosophy - radionuclide containment at the source for all credible plant conditions - has been discussed in numerous publications, but it is perhaps best elaborated in a Preliminary Safety Information Document (PSID) for the 350 MW(t) steam-cycle Modular HTGR (MHTGR) that was submitted to the US Nuclear Regulatory Commission (NRC) in 1987 (PSID 1992). This philosophy has been carried forward for all subsequent MHR designs, including the “reference” NGNP preconceptual design.

This MHR design philosophy has profound implications for the technology development needed to support the design and licensing of such nuclear plants. In particular, it mandates the development and qualification of high-performance coated-particle fuel and the development and scale-up of fabrication processes for its mass production. It also requires a convincing validation of the design methods used to predict radionuclide source terms for normal plant operation and postulated accidents.

This philosophy drives the design and required technology development in other less obvious ways. For example, the requirement to limit fuel temperatures (to <~1600 ^{o}C) during core heatup accidents without active cooling systems lead to the selection of an annular core, an air-cooled RCCS, a limit on thermal power, and the choice of RPV materials among other things. These design selections in turn spawned a large number of DDNs, including those related to the thermal properties of core materials and RPV materials.

2.3.2 MHR Conceptual Designs and Technology Programs

The first GA modular HTGR design developed beyond preconceptual design was the 350 MW(t) steam-cycle MHTGR which was designed to operate on a once-through LEU fuel cycle (PSID 1992). The DDNs for the MHTGR (1989) were systematically identified as part of the detailed functional analysis that was performed for the plant (Section 5.1), and a series of technology development plans were prepared, including a fuel/fission products TDP (Hanson 1987).

The New Production (NP)-MHTGR was based upon this commercial steam-cycle design, but the LEU core was replaced with a core with HEU TRISO-coated, UCO driver fuel and TRISO-coated $LiAl_3O_8$ target particles for tritium production (Lommers 1991). An overall engineering development plan was prepared for the NP-MHTGR (NPR EDP/F 1993) along a complete set of DDNs (Ho 1990) and a series of TDPs, including a fuel/fission product TDP (McCardell 1992).

In the late 1980s, the design of the USDOE MHR evolved from the steam-cycle MHTGR to a direct-cycle Gas Turbine (GT)-MHR, and the module power level was increased first from 350 MW(t) to 450 MW(t) and then to 600 MW(t). These changes dramatically improved the plant economics. A set of DDNs was produced for the commercial 600 MW(t) GT-MHR (GT-MHR DDNs 1996) before the program was terminated by congressional action in 1996. As part of that program, the disposal of spent GT-MHR elements in a Yucca Mountain-like geological repository assessed (Richards 2002), and a supporting technology development plan was also prepared (Hanson 2002)

A design variant of the LEU-fueled, commercial GT-MHR was developed in 1994-1995 for the deposition of surplus US weapons Pu which was designated the Plutonium Consumption (PC)-MHR (1995). An overall engineering development plan was prepared for the PC-MHR (PC-MHR EDP 1995) along with a fuel development plan (Turner 1994) since Pu fuel development was the highest priority task for this project. As part of that program, the disposal of spent PC-MHR elements in a Yucca Mountain-like geological repository assessed (Richards 1994), and a supporting technology development plan was also prepared (Hanson 1995)

The design of the Russian, Pu-burning International GT-MHR (CDR 1997) is largely based upon the PC-MHR design. A comprehensive technology development program (TDPP 2005) is currently in progress, including a Pu fuel development program (RF Fuel Plan 2005).

Preconceptual designs were developed in 2006 for an H2-MHR based upon the SI process (Richards 2006a) and for an H2-MHR based upon high temperature electrolysis (Richards 2006b) under a Nuclear Energy Research Initiative (NERI) contract to General Atomics, Idaho National Laboratory, and Texas A&M University. While these two NERI reports contain some discussion of the required technology development to support the design and licensing of such plants, this present TDP is the first systemic effort to identify all of the DDNs and attendant R&D programs for hydrogen-production MHRs.

When the addressing the subject of the technology development required to support reactor design and licensing, the obvious question is how much technology development is sufficient. This question, especially in the licensing context, is perhaps as much philosophical as it is technical and cannot be answered authoritatively in advance of an actual dialogue with licensing authorities and potential customers. Nevertheless, past precedent may provide some insight. In addition to submitting a PSID for the 350 MW(t) steam-cycle MHTGR, a Regulatory Technology Development Plan (RTDP 1987) and a Probabilistic Risk Assessment (PRA 1988) were also submitted to the NRC. In response, the NRC issued a draft Preliminary Safety Evaluation Report (PSER 1989). In simplest terms, the PSER essentially concluded that the technology development program described in the RTDP was necessary but not sufficient for making the safety case for the MHTGR. Unfortunately, the MHTGR program was terminated prior to reaching a consensus with the NRC on the required scope of the technology development for licensing a steam-cycle MHTGR. Nevertheless, the MHTGR PSER should be carefully considered when finalizing an umbrella TDP for the NGNP.

More recently, a PIRT ("Phenomenon Identification and Ranking Tables") on TRISO particle fuel was prepared for the NRC (Morris 2004). To use the vernacular employed herein, the essential purpose of this PIRT was to define the DDNs (as perceived by the PIRT panel) necessary for licensing TRISO fuel for use in an MHR. From the reactor designer's perspective, the results of this PIRT were frankly disappointing. At face value, it would virtually require a mechanistic first-principles understanding of every phenomenon that might conceivably occur in a TRISO particle in an HTGR environment. The technology program needed to address all of the issues raised by this PIRT would be monumental. In fact, bounding, phenomenologically-based empirical models, when combined with sufficiently large design margins (safety factors), are an acceptable alternative and represent well established engineering practice.

Two additional MHR PIRTs are nearing completion at this writing: one relating to thermal/fluid dynamic phenomena and other to radionuclide transport phenomena. It is anticipated that these three PIRTs will be the topic of considerable discussion during NGNP licensing. They serve to demonstrate the critical importance of early dialogue with the NRC and other licensing authorities (e.g., the USEPA for the H_2 plants) regarding the required technology development for NGNP design and licensing.

2.3.3 Technology Base for H2-MHR Design and Licensing

The technology base for MHR design and licensing derives from five decades of international R&D programs combined with the design, construction and operation of seven He-cooled reactors. Actual reactor operation provides the most credible demonstration of the technology, as well as unresolved issues, and that history is summarized briefly below. The technology base for hydrogen production derives primarily from two sources: (1) the commercial production of inorganic chemicals for more than a century for the SI process, and (2) international development of solid-oxide fuel cells for the past three decades for the HTE process. The status of the various technology development programs is summarized in Section 3.

2.3.3.1 Nuclear Heat Source

2.3.3.1.1 Dragon Reactor Experiment

Gas-cooled reactor design, development and deployment for power generation began shortly after World War II with the CO_2-cooled MAGNOX reactors and Advanced Gas Reactors in Great Britain. The first He-cooled HTGR, the 20 MW(t) Dragon Reactor Experiment, constructed at Winfrith, England, began operation in 1965. The reactor was coupled to steam generators, but the heat was rejected to the environment rather than used for electricity production. A tremendous amount of pioneering R&D, especially related to TRISO fuel, was performed in conjunction with reactor operation (Ashworth 1978).

2.3.3.1.2 Peach Bottom HTGR

The Peach Bottom Atomic Power Station Unit 1 was a 40 MW(e) US prototype HTGR located in eastern Pennsylvania. The heart of the nuclear steam supply system was a helium-cooled, graphite-moderated, 115 MW(t) reactor operating with a 700 °C gas outlet temperature on a thorium-uranium fuel cycle. Peach Bottom (PB) operated successfully for seven years until it was shut down for decommissioning in late 1974 because it had completed its demonstration mission. A comprehensive reactor surveillance program was conducted during plant operation (e.g., Dyer 1977), and an ambitious fuel test element (FTE) program was completed (Saurwein 1982). An extensive and highly successful End-of-Life (EOL) R&D Program, jointly sponsored by USDOE and EPRI, was conducted with the primary goal of generating real-time integral data to validate HTGR design methods with emphasis on reactor physics, core thermal/fluid dynamics, fission product transport, and materials performance, especially performance of the Incoloy 800 used for the steam-generator superheaters (Steward 1978).

2.3.3.1.3 Fort St. Vrain HTGR

The 842 MW(t)/330 MW(e) Fort St. Vrain reactor was the second HTGR built and operated in the USA (FSV FSAR). The reactor operated between 1976 and 1989 for about 875 EFPD (e.g., Baxter 1994). The reactor was rated at 842 MW(t), but it was operated well below that rating for much of its lifetime due to chronic water-ingress problems associated with the water bearings used in the helium circulators (Copinger 2004). For a short time period the reactor operated at 100% design power and achieved a thermal efficiency of 39%. The decommissioning of Fort St. Vrain was completed in 1996 (e.g., Fisher 1998). A reactor surveillance program was performed during plant operation, including fission gas release measurements and examination of two plateout probes (Baxter 1994). A fuel test element program was conducted, but little postirradiation examination (PIE) of the FTEs was performed (SAR 1978). No EOL R&D program was performed at FSV, and a unique opportunity to generate prototypical data for HTGR design verification was lost.

The FSV HTGR had many design features common to prismatic-core MHRs, e.g., graphite moderator, helium coolant, and similar designs for fuel particles, fuel elements, and control rods (e.g., Baxter 1994). Unlike the MHR designs with their steel pressure vessels, the FSV primary coolant circuit was wholly contained within a prestressed concrete reactor vessel (PCRV) with the core and reflectors located in the upper part of the cavity, and the steam generators and circulators located in the lower part. The helium coolant flowed downward through the reactor core and was then directed into the reheater, superheater, evaporator, and the economizer sections of the 12 steam generators. From the steam generators, the helium entered the four circulators and was pumped up, around the outside of the core support floor and the core barrel before entering the plenum above the core to complete the circuit. The superheated and reheated steam was converted to electricity in a conventional, steam-cycle power conversion turbine-generator system.

For FSV, 2448 prismatic fuel elements, 7.1 million fuel compacts containing 26,600 kg of fissile and fertile material in TRISO-coated fuel particles were produced by GA at its now decommissioned, fuel fabrication facility in San Diego. The fissile particle kernels contained fully-enriched uranium carbide and thorium carbide in a ratio of 1 to 3.6. The fertile particle kernels were 100% thorium carbide.

2.3.3.1.4 Arbeitsgemeinschaft Versuchs Reactor

The Arbeitsgemeinschaft Versuchs Reactor (AVR) was a 46 MW(t)/15 MW(e) prototype pebble-bed HTR which began operation in 1968 on a site adjacent to the KFA Juelich national laboratory (now called FZ Juelich). The AVR was used extensively as a test bed for the development and qualification of spherical fuel elements. A broad spectrum of coated-particle types was tested, ranging from the initial core of HEU BISO $(Th,U)C_2$ fuel to high-quality, LEU TRISO UO_2 reload fuel, beginning in 1982. Initially, the AVR operated with a core outlet temperature of 850 ^oC; in 1974, the outlet temperature was raised to 950 ^oC; this temperature increase caused no serious operational problems, but the release rates of Sr and Cs from the older BISO-coated carbide fuel did increase significantly producing rather high plateout inventories in the primary circuit at EOL in 1988. In summary, the AVR proved to be a superb vehicle for the successful development and demonstration of pebble-bed reactor technology, especially for spherical fuel elements (e.g., Gottaut 1990a).

A particularly noteworthy experiment ("HTA-8") was conducted in which 200 unfueled spheres containing meltable wires were inserted into the AVR core. A variation of alloys were included whose melting points span the temperature range from 898 to 1280 °C. These spheres were detected after one passage through the core and recovered. Approximately 20% of the spheres had all wires melted and had therefore seen surface temperatures >1280 °C. A maximum surface temperature of 1150 °C had been predicted during 950 °C operation; the actual fuel temperatures were evidently much higher (Gottaut 1990a).

In a pebble-bed HTR, carbonaceous dust forms from the abrasion of the fuel balls during core transit and recirculation and represents a major transport medium for fission products besides the coolant itself. A number of experiments were conducted in AVR to investigate the effects of dust on radionuclide behavior in the primary circuit (Von der Decken 1990). Of special interest were the VAMPYR-I (Biedermann 1990) and VAMPYR-II (Gottaut 1990b) test facilities that were designed to characterize the transport of condensable radionuclides in the primary circuit of a pebble-bed reactor, including dust effects.

2.3.3.1.5 Thorium-Hoch Temperatur Reaktor

The Thorium-Hoch Temperatur Reaktor (THTR) was a 756 MW(t)/300 MW(e) pebble-bed power plant. The core consisted of 600,000 spherical fuel elements with BISO-coated HEU $(Th,U)0_2$ particles. The plant operated for 423 EFPD, beginning in 1983, until final shutdown in 1988 (Baeumer 1991) because of a combination of factors, including anti-nuclear politics in Germany.

Fuel performance during THTR operation was monitored by on-line measurements of circulating noble gas activity. Initially, the dominant source of gas release was as-manufactured uranium contamination in the fuel-element matrix and OPyC coatings of the BISO particles. An additional source of fission product release developed during reactor operation as a result of mechanical damage to a relatively large number of fuel spheres from the insertion of control rods directly into the pebble bed. A strong correlation was found between the number of damaged fuel element in the core and the circulating gas activity. The upper limits on the fraction of exposed fuel kernels were estimated to be 8×10^{-5} for the entire core and 5×10^{-3} for damaged fuel elements. No EOL R&D program was conducted at THTR.

2.3.3.1.6 High Temperature Engineering Test Reactor

The High Temperature Engineering Test Reactor is a 30 MW(t) test reactor constructed at Oarai, Japan (e.g., Saito 1994). The reactor is designed primarily to investigate nuclear process heat applications. A portion of the nuclear heat (10 MW) is transported from the prismatic core to the secondary cooling loop through an intermediate heat exchanger, constructed of Hastelloy XR (essentially Hastelloy X with a reduced Co content to minimize neutron activation); the remainder of the heat is dumped to a pressurized water cooler. The plant was designed to operate initially with a core outlet temperature of 850 °C and then at 950 °C for process heat applications. Hydrogen production by thermochemical water splitting (sulfur-iodine cycle) will be first investigated. The reactor achieved initial criticality in 1998, full power with a core outlet temperature of 850 °C in 2000, and full power with an outlet temperature of 950 °C in 2004 (Fujimoto 2004).

Fission gas release data from HTTR have been published, but no results for the release of condensable radionuclides are available (there are currently no installed plateout probes). The HTTR is well instrumented for monitoring tritium behavior; however, no results have been published to date. When an SI pilot plant is coupled to the reactor sometime after 2011, there will be a unique opportunity to measure directly the amount of H-3 contamination in the product hydrogen. Presumably, a complete plant mass balance for H-3 will be determined at that time.

2.3.3.1.7 High Temperature Reactor-10

The 10 MW(t) High Temperature Gas-cooled Reactor-Test Module (referred to as the HTR-10) has been constructed at the Institute of Nuclear Energy Technology (INET) of Tsinghua University, Beijing, China, to allow the Chinese to develop an expertise in HTGR technology for potential future applications, including electricity production in direct-cycle plants and process heat applications (e.g., Xu 2002). The HTR-10 is a pebble-bed HTR, based upon German plant and fuel technology. Very little operational data have been published to date.

2.3.3.2 Hydrogen Production

There is a vast chemical engineering literature (e.g., Kirk-Othmer 2004) describing the design of commercial plants for the production of inorganic chemicals, using high-temperature, high-pressure processes. Most of the unit operations involved in the SI process are well established industrial processes. The solid-oxide electrolyzer cells that are the heart of the HTE process are essentially solid-oxide fuel cells operating in reverse; the latter have been under development internationally for more than two decades and are approaching wide-scale commercial deployment for a broad spectrum of applications.

2.4 Key Assumptions/Development Strategy

The DDNs identified in this TDP are based upon the NGNP preconceptual design described in Section 2.2.2. The key assumptions/design selections for purposes of DDN identification are summarized in Table 2-1.

The required plant design and licensing schedule is assumed to be planning Option 2 (“Balanced Risk”) in the PPMP (2006) which requires a 2018 plant startup.

Table 2-2. Key Assumptions/Design Selections for DDN Identification

Parameter	Assumption/Design Selection
NGNP mission	High-efficiency electricity; hydrogen
Reactor type	Modular Helium Reactor (1 module prototype)
Location	INL federal reservation (southern Idaho)
Exclusion Area Boundary	425 m (commercial site requirement)
Off-site accident dose limits	PAGs (1 rem whole body; 5 rem thyroid)
Occupational exposure limits	10% of 10CFR20
Reactor power	550 MW(t) with stretch capability to 600 MW(t)
Reactor pressure vessel material	2¼Cr-1Mo (9Cr-1Mo-V backup)
Core	Prismatic core
Fuel particles (fissile/fertile)	LEU UCO/Natural UCO TRISO (first core LEU UO_2 TRISO)
Fuel compact matrix	Resin matrix
Fuel block	10-row block (FSV)
Graphite – fuel elements	TBD
Graphite – replaceable reflectors	TBD
Graphite – permanent core structures	TBD
Power conversion cycle	Direct, Brayton-cycle gas turbine
Number of primary coolant loops	2 (PCU, IHX)
Primary coolant	Helium
Hot duct material	[IN 617]
Power conversion unit	Vertical, integrated, magnetic bearings (OKBM design)
Turbine blade material	[IN 100]
Recuperator material	[SS 316]
IHX design	PCHE (helical coil backup)
IHX material	IN 617
Core inlet helium temperature	490 °C – 590 °C
Core outlet helium temperature	850 °C – 950 °C
Secondary loop working fluid	Helium
Secondary HTS piping design	parallel
Secondary HTS pipe material	P22 (2¼Cr-1Mo)
Reactor Cavity Cooling System	Air-cooled RCCS
Containment	Vented Low-Pressure Containment
Hydrogen production process	SI, HTE
Hydrogen end-use	TBD
Oxygen end-use	TBD
Tritium contamination of hydrogen	<TBD pCi/liter
Hydrogen plant emissions	“zero discharge“
SI: HI decomposition process	extractive distillation
HTE: SOEC composition	TBD

2.5 Plan Organization and Content

This technology development plan is organized into 11 sections. Section 1 provides a summary of the most important features of the plan.

Section 2 provides a programmatic context and background. It briefly describes the H2-MHR concept and the NGNP prototype, previous MHR conceptual designs and technology programs, the technology base for H2-MHR design and licensing, and the assumptions that had to be made for purposes of DDN identification.

Section 3 describes the status of the technology to support NGNP design and licensing.

Section 4 presents the design requirements that, in large measure, drive the technology development. Those requirements include high performance TRISO fuel, validated RN source terms, operation at 950 ^{o}C, and hydrogen production.

Section 5 describes the Design Data Needs to assure that the design meets the requirements defined in the SRM, in particular, those described in Section 4.

The heart of this technology development plan is Section 6, which describes the technology development activities required to satisfy the DDNs presented in Section 5. It is primarily a critique of the R&D programs planned by the NGNP and NHI programs.

Section 7 describes deficiencies identified in the available test facilities to be used to conduct the R&D described in Section 6.

Section 8 states the Quality Assurance requirements that will be applied to the R&D activities.

Section 9 describes the potential for international collaboration to satisfy the DDNs described in Section 5. International collaboration is highly desirable, if not essential, given the scope and cost of the required R&D and the demanding NGNP schedule.

Section 10 summarizes the current cost and schedule estimates.

Section 11 presents a list of key deliverables that will be prepared in each technology area and compares the planned production dates given in the various technology development plans to when the data are required to support NGNP design and licensing.

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3. TECHNOLOGY DEVELOPMENT STATUS

The current state of key technology areas and their adequacy to support NGNP design and licensing are summarized below. Much of the information is provided by reference.

3.1 Fuel Development and Qualification

The radionuclide containment system for an MHR, which reflects a defense-in-depth philosophy, is comprised of multiple barriers to limit RN release from the core to the environment to insignificant levels during normal operation and a spectrum of postulated accidents (e.g., PSID 1992). As shown schematically in Figure 3-1, the five principal release barriers are: (1) the fuel kernel, (2) the particle coatings, particularly the SiC coating, (3) the fuel element structural graphite, (4) the primary coolant pressure boundary; and (5) the reactor building/containment structure. The effectiveness of these individual barriers in containing radionuclides must be characterized for normal operation and a broad spectrum of postulated accidents.

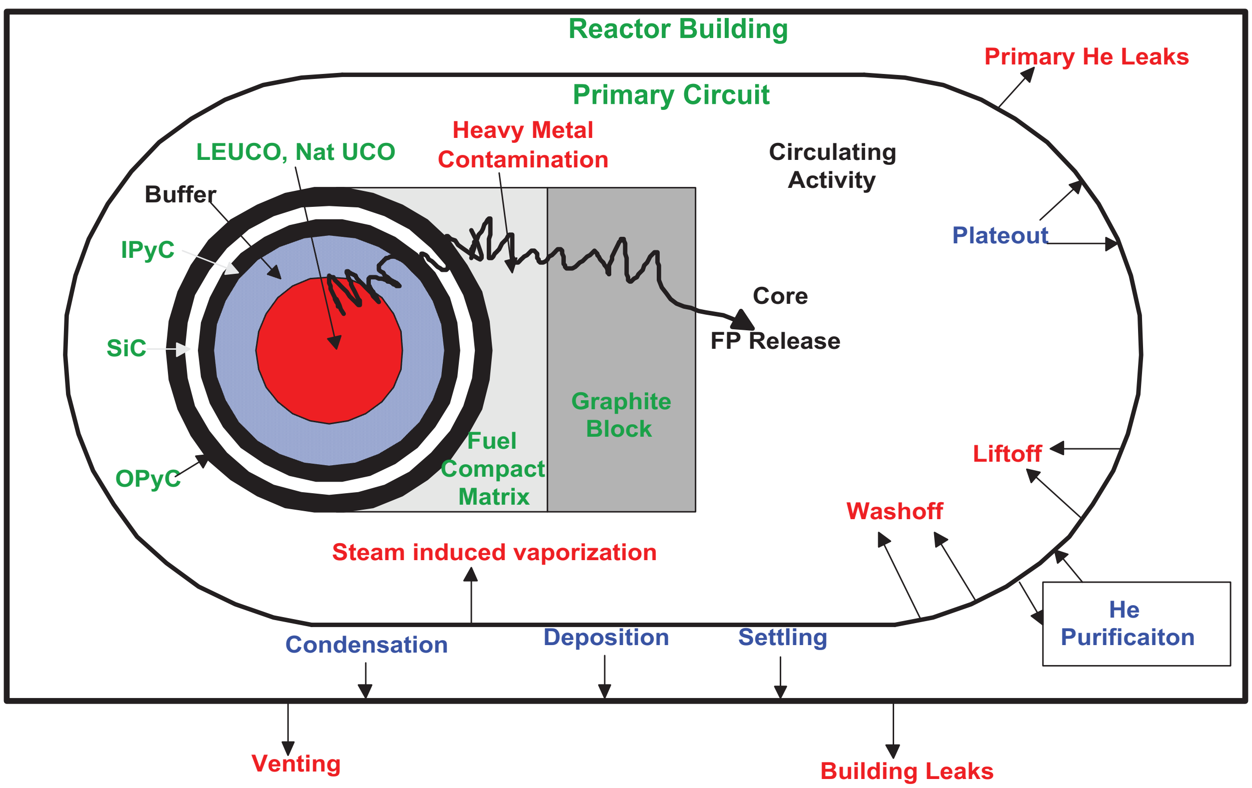

Figure 3-1. MHR Radionuclide Containment System

3.1.1 TRISO Fuel Particles

The most important barrier in the RN containment is the TRISO coating system. The status of TRISO particle technology is summarized in two subsections: (1) fabrication technology and (2) particle performance under irradiation and during accidents.

3.1.1.1 TRISO Particle Fabrication

TRISO particle fuel has been fabricated in many countries throughout the world, irradiated in numerous test capsules, and used as the fuel in power and experimental reactors; thus, the basic processes for fabrication of HTGR fuel are well established. However, the fuel quality requirements for future advanced MHRs are considerably more stringent than for these earlier reactors. The capability of TRISO fuel particles to meet these stringent performance requirements has been demonstrated in Germany for the pebble-bed reactor design, but it has not yet been demonstrated in the USA (or elsewhere) for prismatic core designs. Thus, it is the German quality standard that other national programs (Japan, China, and South Africa) have sought to achieve. Both the German and US programs were discontinued in the early 1990's. The US fuel development program resumed in 2003 with the advent of the AGR program.

3.1.1.1.1 Kernel Fabrication

Ammonia-based precipitation processes, with the two most frequently used variants referred to as "internal gelation" and "external gelation," have undergone extensive development for the production of microspheres containing UO_2, UCO, ThO_2, and mixed systems of U-Th, U-Pu, and Th-Pu as well as pure carbides and nitrides. The UO_2 kernel fabrication process is a mature process that has been used worldwide to fabricate large quantities of kernels. The UCO process is similar to the UO_2 process, but concurrently meeting the kernel density and stoichiometry requirements has proved to be problematic, particularly with increasing kernel size. External gelation processes have been used to manufacture UCO and/or UO_2 kernels at GA in the USA, at Nuclear Fuel Industries in Japan, and at KFA Juelich and HOBEG GmbH in Germany. The internal gelation process has been selected as the reference UCO kernel fabrication process in the US Program. Kernels of 195-μm diameter HEU UCO were fabricated in laboratory scale equipment at BWXT by an internal gelation process for the US NP-MHTGR program in the early 1990's. The feasibility of producing 350-μm and 500-μm diameter UCO kernels by internal gelation has also been demonstrated at BWXT.

The UCO kernel fabrication process begins with preparation of acid deficient uranyl nitrate (ADUN) through the dissolution of uranium oxide (or trioxide) powder in nitric acid. Urea, carbon black, and hexamethylenetetramine are added to the ADUN to make a gelation broth, which is pumped through a vibrating nozzle assembly to produce uniform sized droplets. These droplets form spheres which are immediately dropped into a hot organic liquid, in which the kernels undergo gelation reactions which convert them to a matrix of UO_3 and carbon, with formaldehyde, excess urea, ammonium nitrate, and water dispersed throughout.

The "aged" particles are washed with ammonium hydroxide to remove most of the formaldehyde, urea, and ammonium nitrate, and then dried with warm air to remove some of the water and excess ammonium hydroxide. The kernels are then loaded into a furnace and put through a series of heat treatment steps from about 300 °C to 1800 °C to densify the kernels and to obtain the required kernel chemistry.

The sintered kernels are then screened and tabled to remove kernels that are oversized, undersized, or non-spherical. Tabling involves passing the kernels over a vibrating, flat, inclined surface where non-spherical particles, which do not roll like the spherical particles, tend to take a side-ways path off the table and are thereby separated from the spherical particles, which roll down the table. Finally, multiple batches of kernels are mixed together and homogenized to form a kernel “composite” for the purpose of QC acceptance testing and to improve the uniformity of the intermediate product for the next fabrication step (i.e., coating). Performing QC acceptance testing on composites, as opposed to individual batches, allows for a substantial reduction in QC costs.

3.1.1.1.2 Coated Particle Fabrication

As previously shown in Figure 2-5, the TRISO-particle coating system consists of multiple layers of pyrocarbon sandwiched around a SiC layer. The coating layers are produced by chemical vapor deposition (CVD) in a fluidized bed furnace. The pyrocarbon layers are deposited by the thermal decomposition of acetylene or a combination of acetylene and propylene. Argon is used as the fluidizing gas in pyrocarbon coating. The SiC layer is deposited by thermal decomposition of methyltrichlorosilane. Hydrogen is used as the fluidizing gas in SiC coating and as the carrier gas for the methyltrichlorosilane. In the USA, the coating layers have historically been deposited in a sequence of coating steps in one or more coating furnaces with unloading and handling of the intermediate product. However, coating experience in Japan and Germany has shown that the quality level of the coated particles is better (i.e., the fraction of particles with defective SiC coatings is minimized) if all of the coating layers are deposited in an uninterrupted sequence in a single coating furnace (“once through coating”). Thus, uninterrupted sequential coating is considered the reference coating process for future coated particle fabrication in the USA

Proper fluidization of the particle bed, which is essential in order to produce high quality coatings, is a function of the total gas flow, the mass and volume of the particles, the diameter of the furnace, and the design of the gas distributor. Given proper fluidization of the particles, the independent variables that have the greatest influence on the microstructural properties of the coatings are the temperature of the fluidized bed and the active coating gas fraction. (The active coating gas fraction is the coating gas flow divided by the sum of the fluidizing gas flow and the active coating gas flow.) The uniformity of the coating environment is also very important. If the coating gas concentrations in different areas of the coater are substantially different, and if particles are not adequately fluidized to randomize their movement in the coater, then wide variations in coating rate and coating properties will be observed within a coated particle batch. This is highly undesirable so it is important that the coating furnace be properly designed. Lack of coating uniformity within the coater is a major quality issue that must be overcome in scaling up from small-diameter laboratory coaters to large commercial coaters.

As is the case with the kernels, the coated particle batches are screened and tabled to remove particles that are oversized, undersized, or non-spherical, and then combined into composites for QC acceptance testing and to improve the uniformity of the coated particles delivered to compact fabrication.

3.1.1.1.3 Compact Fabrication

Extensive experience in the production of cylindrical fuel compacts was gained at GA during mass production of FSV fuel. In the FSV compact fabrication process, a thermoplastic matrix composed primarily of graphite powder and petroleum pitch (as the binder) was injected into a bed of particles in a mold to form a compact. The “green” compacts from the compact-forming operation were then heated at about 900 °C in flowing argon to carbonize the matrix and then heated at up to about 1750 °C in flowing argon to drive out impurities and to partially graphitize the matrix. The thermoplastic matrix injection process was developed and used for FSV fuel production primarily because of its suitability for making compacts with high particle packing fractions. However, this process has a number of drawbacks. The injection process requires compaction of the bed of particles, which is a potential source of coating breakage. Also, the compacts must be supported by alumina powder during carbonization to prevent them from losing their shape. Furthermore the petroleum pitch and alumina powder are sources of impurities that are known to attack SiC coatings.

The fuel quality requirements for current MHR designs are much more stringent than for FSV, so the compact fabrication process must be capable of reducing the level of heavy-metal contamination and defective particles in compacts by more than an order of magnitude compared to the levels demonstrated during FSV fuel production. To achieve this capability, a compacting process improvement program was conducted at General Atomics in 1995-1996. After process changes were made to reduce impurity levels, fuel compacts that met commercial GT-MHR product quality specifications with large margins were fabricated.

While GA developed and utilized a thermoplastic-matrix based compacting process because of the high fuel particle packing fraction requirements for FSV, the rest of the international HTGR community focused on thermosetting-matrix-based compacting processes. Successful compacting in which a synthetic thermosetting resin was used as the binder was demonstrated for fuel elements containing overcoated fuel particles for the pebble bed reactor programs in Germany and in China. Annular fuel compacts have been developed in Japan using a similar process.

In the overcoating process, TRISO-coated particles are loaded into a drum-like overcoating vessel and rotated while matrix powder and methanol are added. The methanol wets the particles and matrix powder and causes the matrix powder to build up a coating layer on the particles. The amount of graphite powder and methanol and the rate of addition are carefully controlled to achieve the desired overcoating thickness. The overcoated particles are dried at about 80 °C and screened and tabled to remove undersized, oversized, and non-spherical particles.

3.1.1.1.4 Quality Control Techniques

QC methods for TRISO fuel particles are well established and have been used for relatively large-scale fuel production in the USA and in Germany. However, the fuel product specifications that have been used historically for TRISO fuel are not sufficiently comprehensive to ensure the required irradiation performance of the fuel; consequently, a combination of product specifications and process specifications have been used to ensure the necessary product quality. New QC methods for characterization of the stoichiometry of individual UCO kernels and IPyC coating permeability and anisotropy must be developed, and enhanced methods for characterizing SiC microstructure and defects may also be needed, but this has not been established conclusively. Furthermore, many of the existing QC methods employ 1970's technology and are too time consuming and costly to support economical large-scale fuel production. Consequently, there is a need to develop low-cost automated nondestructive methods capable of high throughput rates and of providing near real-time feedback to the fuel fabrication processes.

3.1.1.2 TRISO Particle Performance

During the past four decades, a number of mechanisms have been identified - and quantified - which can compromise the capability of the coated fuel particles to retain radionuclides (i.e., functional failure of the coated particle). A considerable number of documents have been prepared on the topic of coated particle failure mechanisms. TECDOC-978 (1997) provides a good summary along with an extensive bibliography.

The following failure mechanisms have been identified as capable of causing partial or total failure of the TRISO coating system under irradiation and/or during postulated accidents; these mechanisms are shown schematically in Figure 3-2. Phenomenological performance models, typically inspired by first principles and correlated with experimental data, have been developed to model each of these mechanisms.

1. Coating damage during fuel manufacture, resulting in heavy metal contamination on coating surfaces and in the fuel compact matrix.
2. Pressure vessel failure of standard particles (i.e., particles without manufacturing defects).
3. Pressure vessel failure of particles with defective or missing coatings.
4. Irradiation induced failure of the OPyC coating;
5. Irradiation induced failure of the IPyC coating and potential SiC cracking;
6. Failure of the SiC coating due to kernel migration in the presence of a temperature gradient.
7. Failure of the SiC coating caused by fission product/SiC interactions.
8. Failure of the SiC coating by thermal decomposition.
9. Failure of the SiC coating due to heavy-metal dispersion in the IPyC coating.

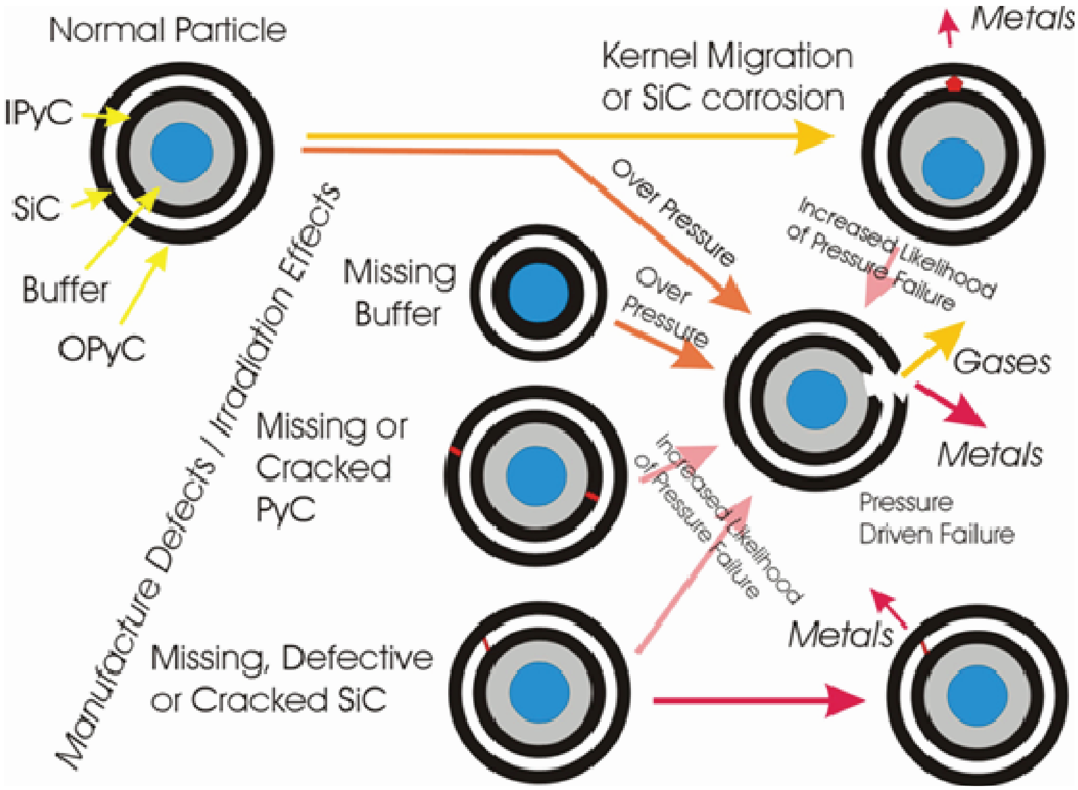

Figure 3-2. TRISO Particle Failure Mechanisms

The first mechanism listed above – as-manufactured heavy-metal contamination - is not an in service failure mechanism *per se* but rather an extreme case of as-manufactured coating defects whereby trace amounts of heavy metal are not encapsulated by a single intact coating layer (analogous to "tramp uranium" in LWR fuel). Modern fuel product specifications only allow small fractions of HM contamination (~10^{-5} is typical); nevertheless, it is an important source of fission product release.

3.1.2 Advanced Fuel Particles

A number of candidate advanced coated-particle designs have been explored which appear to promise superior high temperature performance compared to conventional SiC-based TRISO particles (e.g., Hanson 2003). Typically, these advanced particle designs have been fabricated in small quantities in laboratory-scale equipment and subjected to varying degrees of exploratory testing, including out-of-pile tests, irradiation tests, PIE, and postirradiation heating (PIH) tests. Two promising advanced particle designs appear to be more mature than the others (at least based upon information published in the open literature): (1) TRISO-coated UO_2^* (referred to as "UO_2-star," a conventional UO_2 kernel with a thin ZrC overcoat) and (2) "TRIZO"-coated (ZrC replacing SiC) UCO. The available data on UO_2^* and on ZrC coatings have been reviewed previously (e.g., in Section 7 of TECDOC-978 1997).

It should not be surprising that the two leading advanced fuel designs represent incremental changes in the conventional FRG and US particle designs, respectively. The UO_2^* particle, of which there are two variants, is essentially a modification of the standard FRG TRISO-coated UO_2 particle. The only design change is the addition of ZrC to the particle: either as a thin ZrC coating applied over a thin PyC seal coat on the UO_2 kernel (referred to as UO_2^*-C) or co-deposited with the porous PyC buffer layer (referred to as UO_2^*-B). UO_2^* particles, especially the UO_2^*-C variant, appear to perform far better than conventional TRISO-coated particles (e.g., Ag-110m is completely retained at 1500 ^oC for 10,000 hours). The TRIZO particle is the standard LEU UCO particle with the SiC coating replaced by a ZrC coating. ZrC coatings are more thermally stable than SiC and are not degraded by palladium attack at high temperatures (>~1400 ^oC). A plan to develop and qualify these two advanced fuels has been prepared (Hanson 2003), but it has not been funded to date.

Other more exotic advanced fuels have been, or are being investigated for use in various reactor designs. These include, for example, new fuels for fast gas-reactors in which pellets of fuel are coated with a material such as titanium nitride as an alternative to graphite.[5] Also, CVD niobium carbide-coated uranium oxide fuel and binary carbide fuels of (U, Zr)C were investigated in the KIWI and NERVA nuclear rocket propulsion programs in the 1960's.[6] Uranium bearing, solid-solution tri-carbide fuels such as (U, Zr, X)C, where X = Nb, Ta, Hf, or W have been investigated at the Innovative Nuclear Space Power and Propulsion Institute at the University of Florida for advanced space power and propulsion applications.[7] According to this reference, the presence of non-uranium carbides in the tri-carbide fuel allows for gradient coating of fuel pellets with refractory metal carbides for fission product containment, and no additional coating is necessary as with earlier graphite matrix and composite fuels. Although these advanced fuels are intriguing, it should be noted that there is nothing in the literature concerning any research and development that has been aimed toward the use of these fuels in a VHTR.

Considerable research has been done and continues in the development of inert matrix fuels (IMF) to facilitate the burning of weapons-grade plutonium and commercial-reactor plutonium (and higher actinides) in LWRs. If this burning of fissile actinides is to be accomplished in LWRs without the inclusion of uranium (for non-proliferation considerations, etc.), then inert materials which act as diluents must be added to reduce the fission rate density and the effective burnup. A broad spectrum of ceramic materials has been evaluated for use (e.g., Journal of Nuclear Materials 1999) diluents in IMF in LWRs, and some may have application in coated-particles fuels as well. Although TRISO-coated particles with highly enriched uranium and highly enriched plutonium have been successfully irradiated to high burnup (>70% FIMA), the International GT-MHR program is also evaluating a TRISO-coated PuO_x kernel diluted with carbon or zirconium as a backup to the reference, TRISO-coated, pure fissile PuO_x particle (RF Fuel Plan 2005).

[5] http://www.cea.fr/gb/publications/Clefs45/clefs45gb/clefs4521a.html
[6] http://www.fas.org/nuke/space/c04rover.htm
[7] http://www.inspi.ufl.edu/tricarbide.pdf

3.2 Radionuclide Source Terms

The status of the technology currently available to predict radionuclide transport in HTGRs has been described previously (e.g., TECDOC-978 1997, Martin 1993, Hanson 2004, etc.); TECDOC-978 is still the best single reference even if it is a decade old. The purpose of this section is not to provide another comprehensive description of this technology but rather to provide a brief qualitative summary and a bibliography. The information is summarized in three subsections. Subsection 3.2.1 describes the fuel performance and radionuclide transport codes used for reactor design and safety analysis. Subsection 3.2.2 summarizes the material property database from which the input correlations for these codes were derived. Subsection 3.2.3 describes the previous efforts to validate these codes by comparing code predictions with the observed radionuclide transport behavior in operating reactors and test facilities.

3.2.1 Radionuclide Transport Codes

The design methods currently available to predict the various fuel performance and fission product transport phenomena in HTGRs are summarized in Table 3-1 (Hanson 2007). At a minimum, these computational methods will be required to predict the radionuclide source terms during NGNP conceptual and preliminary designs; they will be need to upgraded and validated prior to completion of Final Design.

3.2.2 Component Models and Material Property Data

The reference GA component models and material property correlations are contained in the Fuel Design Data Manual, Issue F (FDDM/F, Myers 1987). The FDDM/F has several notable limitations; in particular, it presents models and correlations along with extensive references, but it does not include the experimental data from which they were derived. In recognition of the above limitations, Martin of ORNL prepared a compilation in 1993 which collected the GA models and the supporting database under a single cover (Martin 1993).

3.2.2.1 Radionuclide Transport in Reactor Core

As with fuel particle failure, a number of mechanisms have been identified - and quantified – which govern the transport of radionuclides in HTGR core materials, and a large number of documents have been prepared on the topic (e.g., Nabielek 1974, TECDOC-978 1997, Martin 1993, Hanson 2004, etc). Especially notable is the 1974 Dragon Project Report DP-828, Part III, by Nabielek which provides a comprehensive set of transport models along with analytical solutions for many bounding cases; this report remains as useful today as it was three decades ago despite the development of numerical methods for predicting fission product transport. Once again, TECDOC-978 (1997) provides a good summary of radionuclide transport phenomena in HTGR core materials along with an extensive bibliography.

The transport of radionuclides from the location of their birth through the various material regions of the core to their release into the helium coolant is a relatively complicated process. The principal steps and pathways are shown schematically in Figure 3-3. (For a pebble-bed core, those steps related to transport across the gap between the fuel compact and the fuel-element and transport in the fuel-element graphite are eliminated.) Also for certain classes of radionuclides, some steps are eliminated (e.g., noble gases are not diffusively released from intact TRISO particles, but noble gases are not significantly retarded by the compact matrix or fuel-element graphite.

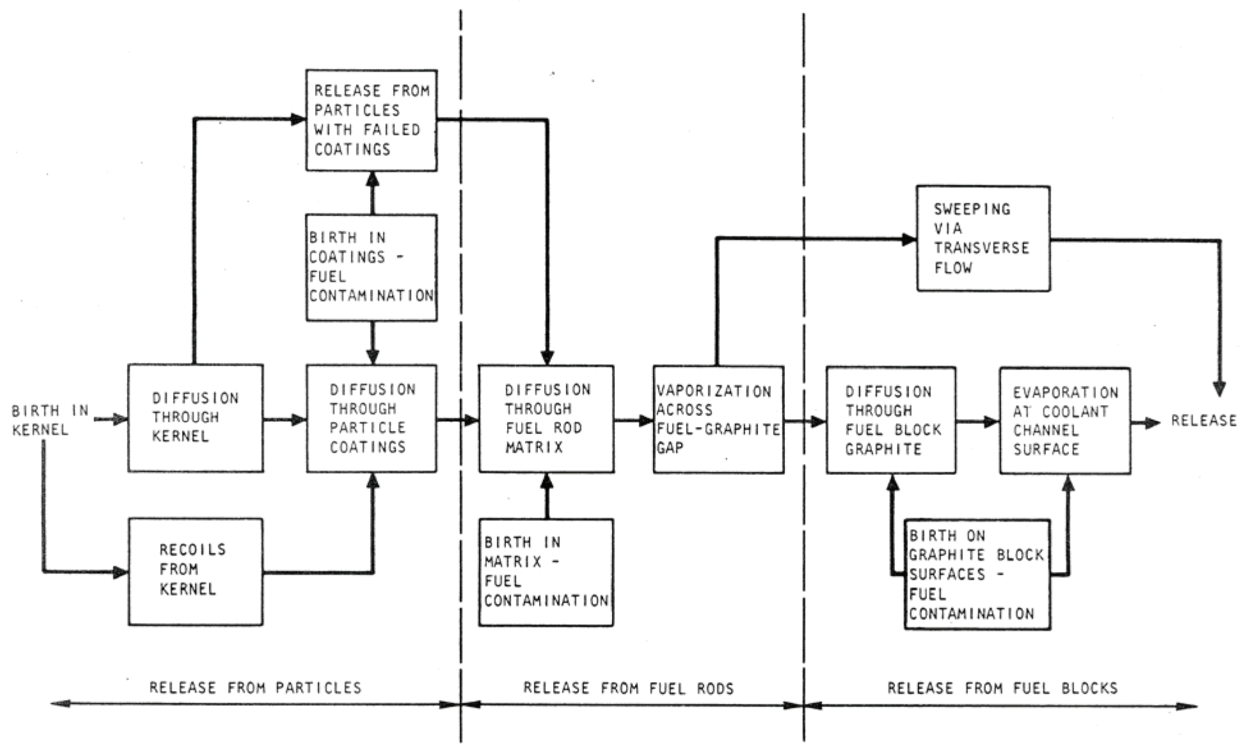

Figure 3-3. Principal Steps in Radionuclide Release from an HTGR Core

The two dominant sources of fission product release from the core are as-manufactured heavy metal contamination and particles whose coatings fail in service. The latter source can be subdivided into (1) coating failure during normal operation and (2) incremental coating failure during core heatup accidents. In addition, certain volatile fission metals (notably Ag) can, at sufficiently high temperatures and long times, diffuse through the SiC coatings of intact TRISO particles.

Radionuclide transport must be modeled in the fuel kernel, particle coatings, fuel-compact matrix, and fuel-element graphite. While the actual radionuclide transport phenomena in an HTGR core are complex and remain incompletely characterized after four decades of modeling efforts, the basic approach remains unchanged; radionuclide transport is essentially treated as a transient solid-state diffusion problem with various modifications and/or additions to account for the effects of irradiation and heterogeneities in the core materials.

The radiologically significant fission gases (e.g., 2.8-hr Kr-88) are typically short-lived, the production rates and release rates reach equilibrium quickly, and the steady-state, core service conditions change relatively slowly by comparison; consequently, steady-state approximations are typically used in core design and analysis methods for predicting fission gas release during normal operation (Haire 1974). The release of fission gases from heavy-metal contamination and from fuel kernels under irradiation is typically expressed in terms of the release rate-to-birth rate ratio (R/B); at steady-state, the R/B ratio is numerically equal to the fractional release. The gas release models give the R/Bs from contamination and failed particles as a function of chemical element, isotope half life, temperature, and burnup (Myers 1987). These functional dependencies are determined experimentally.

The models for core heatup conditions are necessarily transient because of the large changes in temperature. The experimental data for gas release from failed particles and heavy-metal contamination suggest a two-component model with a fraction of the inventory being rapidly released and a fraction being slowly released (Myers 1978).

In contrast to the fission gases, the key fission metals (e.g., 30-yr Cs-137) are typically long-lived, and transient analysis methods are necessary for both normal and core heatup conditions (Alberstein 1975). In fact, the same models and material property correlations are used for both normal operations and accidents. The transport of fission metals through the kernel is modeled as a transient diffusion process. (Nabielek 1974) provides a number of analytical solutions for bounding conditions (e.g., constant power and constant temperature), but the conservation equations (based upon Fick's 2nd law of diffusion) are typically solved numerically with appropriate boundary and interface conditions which vary depending upon whether a bare kernel or an encapsulated kernel in an intact coated particle is being modeled. This approach is taken in TRAFIC-FD (Tzung 1992a) and COPAR-FD (Tzung 1992b).

The transport of the volatile fission metals, including Ag, Cs, Sr, and Eu, in the PyC and SiC coatings is also treated as a transient Fickian diffusion process. In this case, the geometry is a spherical shell, and the interface conditions between the layers are assumed to be described by partition factors (e.g., Tzung 1992b).

Fission metal transport in the fuel-compact matrix and fuel-element graphite is again modeled as a transient Fickian diffusion process: the transient diffusion equation for cylindrical geometry is solved with an evaporative boundary condition. It is assumed that sorption equilibrium prevails in the gap between the fuel compact and the fuel hole surface of the fuel block. At equilibrium, the vapor pressure in the helium-filled gap and solid-phase concentration on the fuel-compact surface are uniquely related to one another by a sorption isotherm which is determined experimentally. This approach is taken in TRAFIC FD (Tzung 1992a).

Several different sorption isotherms have been derived by making various assumptions about the potential energy distributions of the sorption sites which lead to different functional dependencies between the gas-phase partial pressure and the surface concentration; however, for the sorption of fission products on core materials, the experimental data are generally correlated with a simple Henrian isotherm (linear dependence) for low sorbate concentrations and with a Freundlich isotherm (exponential dependence) at higher sorbate concentrations (Myers 1987).

Sorption isotherms for Cs, Sr and Ag have been measured for a variety of nuclear graphites and matrix materials; the data are summarized in TECDOC-978. Measurements have been made on both unirradiated and irradiated materials. For matrix materials with a high content of amorphous carbon, irradiation has little effect on the sorptivity; however, for highly graphitic materials, the Cs and Sr sorptivities are observed to increase with increasing fast neutron fluence. Apparently, neutron irradiation of the crystalline component causes damage which serves to create additional sorption sites. Consequently, the sorption isotherms have been modified to include a fast fluence dependence which is fit to the experimental data.

At the coolant boundary, the mass flux from the surface into the flowing coolant is given by the product of a convective mass transfer coefficient and a concentration driving force which is the difference between the desorption pressure (expressed as a volumetric concentration) and the “free stream” or mixed mean concentration in the coolant. For prismatic fuel elements, the mass transfer coefficient is calculated from an empirical correlation for the Sherwood number. The reference GA correlations for predicting convective mass transfer coefficients for forced convection and free convection are cataloged in the PADLOC code (Hudritsch 1981). In general, the Sherwood number is given as functions of the Reynolds, Schmidt, and Grashof numbers.

3.2.2.2 Radionuclide Transport in Primary Circuit

A variety of plateout models have been developed internationally (e.g., TECDOC 978 1997, Hanson 2002, etc.). In general, they are derived by solving mass balance equations for a gas contaminated with radionuclides flowing through a conduit with various boundary and interface conditions relating the concentration of the radionuclide in the coolant to its concentration on the fixed surface.

The key material property correlations used in calculating the deposition of condensable fission products in the primary circuit are convective mass transfer correlations and sorption isotherms; these isotherms predict the equilibrium surface loading as a function of partial pressure and surface temperature. The reference GA sorption isotherms and the experimental data from which they were derived are summarized in (Myers 1981, Myers 1984).

The circulating and plateout activities in the primary coolant circuit are potential sources of environmental release in the event of primary coolant leaks or as a result of the venting of primary coolant in response to overpressuring of the primary circuit (e.g., in response to significant water ingress in a steam-cycle plant). The fraction of the circulating activity lost during such events is essentially the same as the fraction of the primary coolant that is released, although the radionuclide release can be mitigated by pumpdown through the helium purification system if the leak rate is sufficiently slow.

A fraction of the plateout may also be reentrained, or “lifted off,” if the rate of depressurization is sufficiently rapid. The amount of fission product liftoff is expected to be strongly influenced by the amount of particulate matter (“dust”) in the primary circuit as well as by the presence of friable surface films on primary circuit components which could possibly spall off during a rapid depressurization. Simple empirical models have been used to correlate measured liftoff fractions with gas dynamic parameters (e.g., TECDOC-978 1997). GA has traditionally employed an empirical shear ratio[8] (SR) model for correlating liftoff data (Myers 1986). While there are many valid criticisms of this simplistic model, the currently available liftoff data do not appear to justify a more complicated one.

Other mechanisms which can potentially result in the removal and subsequent environmental release of primary circuit plateout activity are “steam-induced vaporization” and “washoff.” In both cases, the vehicle for radionuclide release from the primary circuit is water which has entered the primary circuit. In principle, both water vapor and liquid water could partially remove plateout activity (Myers 1986). However, even if a fraction of the plateout activity were removed from the fixed surfaces, there would be environmental release only in the case of venting of helium/steam from the primary circuit. For all but the largest water ingress events the pressure relief valve does not lift. Moreover, the radiologically important nuclides such as iodine and cesium are expected to remain preferentially in the liquid water which remains inside the primary circuit. The probability of a large water ingress in a H2-MHR with its IHX is much lower than for a steam-cycle plant.

The POLO code (Bolin 1989) is used to calculate the integral fission product liftoff in the primary coolant circuit during a depressurization transient. POLO contains the above empirical correlation which gives the fractional liftoff as a function of shear ratio. The initial plateout distribution is obtained from the PADLOC code, and the shear ratio distribution in the primary circuit as a function of time and space is obtained from a transient thermal/fluid dynamics code. POLO then integrates the transient shear ratio distribution over the initial plateout distribution to predict the cumulative release from the primary circuit. (POLO also contains a simple washoff model for the analysis of wet depressurization transients.)

3.2.2.3 Radionuclide Transport in Reactor Building

The reactor building/containment structure is the fifth barrier to the release of radionuclides to the environment. Its effectiveness as a release barrier is highly event-specific. The vented low pressure containment (VLPC) may be of limited value during rapid depressurization transients; however, it is of major importance during higher risk, core conduction cool-down (CCCD) transients during which forced cooling is unavailable. Under such conditions, natural removal mechanisms occurring in the VLPC, including condensation, fallout and plateout, serve to attenuate the release of condensable radionuclides, including radiologically important iodines, by at least an order of magnitude.

Design methods have been developed to predict radionuclide transport in the VLPC during various postulated accident scenarios. Certain computer codes, including MACCS and MELCOR, that were developed by the water-reactor community, have been acquired and are used for some applications, including the calculation of off-site accident doses. The codes require extensive input, including the transient radionuclide release rates into the VLPC, the physical and chemical forms of these radionuclides, environmental conditions within the VLPC

[8] The ratio of the wall shear stress during depressurization transients to that during normal steady-state operation.

(temperatures, relative humidity, etc.), and correlations (e.g., sorption isotherms, etc.) describing the interactions of the various radionuclides with the exposed surfaces within the VLPC (e.g., metals, painted surfaces, concrete, etc.).

No direct measurements have been made of radionuclide removal from contaminated helium by condensation, settling, and plateout under the conditions expected in the VLPC during a core heatup transient. There is an extensive LWR and CANDU database on the behavior of radionuclides in water-reactor containment buildings, and major experimental programs are in progress to further characterize the behavior of radionuclides in LWR containment buildings (e.g., the international PHEBUS test program in France). Some of these LWR data, especially those that relate to radionuclide partitioning between steam and liquid phases in steam-water mixtures, may be applicable to radionuclide behavior in VLPCs.

An evaluation was recently made to determine the extent to which the extensive international database for radionuclide transport in water-reactor containments might be applied to refine and to independently validate the design methods used for predicting radionuclide transport in the VLPCs of modern MHRs (Hanson 2007b). In summary, the experimental water-reactor database for radionuclide transport in containment buildings was judged to be of limited value for refining and independently validating the design methods used to predict radionuclide transport in VLPCs because the radionuclide concentrations and physical and chemical forms in the two systems are too different. Thus, an experimental program needs to be planned and conducted to characterize radionuclide transport under the conditions predicted for the VLPC during specific MHR accident scenarios.

3.2.2.4 Tritium Behavior in HTGRs

A radionuclide containment issue of special interest for the H2-MHR is containment of tritium (Hanson 2006). Tritium will be produced in a H2-MHR by various nuclear reactions. Given its high mobility, especially at high temperatures, some tritium will permeate through the IHX and SI process vessels, contaminating the product hydrogen. This tritium contamination will contribute to public and occupational radiation exposures; consequently, stringent limits on tritium contamination in the product hydrogen are anticipated to be imposed by regulatory authorities. Design options are available to control tritium in a VHTR, but they can be expensive so an optimal combination of mitigating features must be implemented in the design.

The following sources of tritium production have been identified, primarily from early surveillance programs at operating HTGRs (steam-cycle plants), and they can be reasonably quantified for a H2 MHR: (1) ternary fission, (2) neutron activation of He-3 in the primary He coolant, (3) neutron activation of lithium impurities in fuel-compact matrix and core graphite, and (4) neutron capture reactions in boron in control materials. Ternary fission will be the dominant source of tritium production, but much this tritium will be largely retained in the TRISO-coated fuel particles. He-3 activation will generate a relatively modest fraction of the total tritium production in the reactor; however, since it is born in the primary coolant, it will likely be the dominant source of tritium in the primary helium and, hence, the dominant source of product contamination as well.

Tritium strongly chemisorbs on irradiated nuclear graphite at elevated temperatures. Consequently, a large fraction of the tritium entering the primary helium will be sorbed on the huge mass graphite in the core. In operating HTGRs, including Fort St. Vrain, the core graphite was a far more important sink for tritium removal than the helium purification systems. However, a large fraction of this stored tritium can be released if water is introduced into the primary coolant (a low-probability event for a VHTR with an IHX).

Tritium will permeate through the heat exchangers and process piping in an H2-MHR and will contaminate the product hydrogen. Surface films will play a critically important role in establishing the in-reactor, tritium permeation rates. Oxide films can reduce H-3 permeability by orders of magnitude. However, normal plant operating transients (e.g., startup/shutdown, etc.) may compromise film integrity and result in increased H 3 permeation rates.

Design methods are available to estimate H-3 production, distribution, and release, but they are rudimentary and characterized by large uncertainties (Myers 1987, Hanson 2006a). The current design methods appear adequate for Conceptual Design, but they will need to be upgraded for Preliminary Design and independently validated prior to completion of Final Design. Some technology development will be necessary to provide the basis for these design methods improvements and validation.

3.2.3 Design Methods Validation

A number of attempts have been made to validate the design methods described above by comparing code predictions with integral radionuclide transport data from fuel irradiation capsules, in-pile fission product transport loops, and operating HTGRs. The results have been summarized previously (TECDOC-978 1997, Hanson 2002, Hanson 2004); those summaries contain extensive bibliographies that will not be repeated here.

3.2.3.1 Radionuclide Release from Reactor Core

The results of such comparisons are often ambiguous because the measurements are quite integral. The measured radionuclide releases from a fuel irradiation capsule or, even more so, from a reactor core represent the integral of multiple sources (HM contamination, failed particles, etc.), the fractional releases of radionuclides from each of the sources, and, in case of metallic fission products, the degree of attenuation by the compact matrix and fuel-element graphite. Moreover, large variations in fuel burnup, fast fluence, and especially temperature are common in irradiation capsules and inevitable in a reactor core. To model such systems, the sources (e.g., coating failure rates, etc.) and the radionuclide transport through each release barrier must be predicted. Consequently, when there is good apparent agreement between predictions and measurements, one can not in general be certain that there were not compensating errors. Likewise, when there are significant discrepancies, the cause(s) may be difficult to isolate.

For some experimental programs, this ambiguity has been removed by including a known fission product source. For example, laser-failed particles, bare kernels, and “designed-to-fail” particles (standard fuel kernels with a thin PyC seal coat) have been seeded into fuel compacts to provide a known source. Such seeded compacts have been irradiated in capsules and in-pile loops.

The fission gas release from irradiation capsules containing LEU UCO/ThO_2 fuel is generally predicted to within a factor of about five. However, these capsules operated dry, so the hydrolysis model was not tested independent of the data from which it was derived. Moreover, there is the aforementioned inherent ambiguity in these data since the fuel failure fraction is not known with high accuracy independent of the gas release data (certain PIE measurements do provide some independent indication of the particle failure fraction).

The validity of the methods for predicting fission metal release from the core during normal operation have been assessed by applying the reference design methods to predict the observed metal release in irradiation capsules: e.g., SSL-1, SSL2, Idylle 03, and R2 K13; in-pile loops: e.g., four CPL2 loops, and COMEDIE BD-1; and in operating HTGRs: e.g., Peach Bottom Core 2 and FSV. Most of the available data are for the Cs isotopes with a small amount of Ag and Sr data. The releases of fission metals were often underpredicted by large factors, and in some cases, by more than a factor of 10. The cause of the underpredictions is ambiguous because the SiC defect fractions and the particle failure fractions are typically not well known. However, there is circumstantial evidence that the transport across the fuel compact/fuel element gap and the transport in the graphite web are not properly modeled.

3.2.3.2 Radionuclide Transport in Primary Circuit

The validity of the methods for predicting plateout distributions in the primary circuit during normal operation have been assessed by applying the reference design methods to predict the observed distributions in out-of-pile loops: e.g., BMI loop and GA deposition loop; in-pile loops: e.g., four CPL2 loops, and COMEDIE SR-1 and BD-1; and in operating HTGRs: e.g., Peach Bottom and Fort St Vrain. Most of the available data are for the Cs isotopes with some Ag, I and Te data. The results were quite variable: in some cases excellent agreement was observed (e.g., Cs plateout in Peach Bottom); in other cases, very large discrepancies (>>10x) were observed (e.g., Ag plateout in COMEDIE SR-1). In general, the reason for the discrepancies appeared to be inadequate sorption isotherms to describe the sorptivities of volatile radionuclides (I, Cs, Ag) on primary circuit alloys. This behavior can be illustrated two prominent examples: Peach Bottom and COMEDIE BD-1.

3.2.3.2.1 Plateout in Peach Bottom

The plateout distributions of gamma-emitting nuclides in the Peach Bottom primary circuit at end-of-life were determined by *in-situ* scanning (Hanson 1976). The dominant gamma-emitting nuclides were Cs-137 and Cs-134; their distributions were similar. Radioassay of the destructively removed samples confirmed the specific cesium activities determined by the *in-situ* scanning; Sr-90 was also measured, but the specific strontium activity was about 1/1000 that of cesium. Neutron activation analysis of leach solutions failed to detect any I-129 or Te-126.

The major difficulty in predicting cesium plateout in Peach Bottom was the choice of isotherms to describe the sorptive capacity of the surface. The hot duct cladding was constructed of

SS-304 for which sorption data are available. For SS-304 the oxidation state of the surface has a large effect with oxidation favoring increased sorption. Another complication was that all exposed surfaces in the PB primary circuit were covered with a carbonaceous deposit produced by cracking of lubricating oil that leaked periodically from a purified helium transfer compressor into the primary circuit. This carbon deposit was likely a significant sink for cesium. Conceivably, the plateout surfaces may be more appropriately characterized as carbonaceous rather than metallic. Since the cesium sorptivity of this carbon deposit was unknown, it was evaluated parametrically by assuming that the deposit had a sorptivity ranging from that of graphite to petroleum-derived, fuel-compact matrix.

In summary, the experimentally observed cesium plateout distribution in Peach Bottom can be predicted almost exactly, providing appropriate sorption isotherms are employed (Hanson 1976). However, the observed sorption behavior is consistent with either assuming that the primary cesium sink is a relatively oxide-free SS-304 surface or assuming that the carbon deposit has a cesium sorptivity intermediate to that of graphite and matrix. These assumptions seem equally feasible; in reality, both probably contributed to the total sorptive capacity of the surface. With the exception of the hot duct which experienced surface temperatures of ~700 ^{o}C, cesium deposition throughout the primary circuit was likely mass transfer controlled; moreover, the deposition profiles indicate that cesium was transported primarily in atomic form despite the ubiquitous presence of carbonaceous dust in the primary circuit.

3.2.3.2.2 Plateout and Liftoff in COMEDIE BD-1

The COMEDIE BD-1 in-pile loop test was conducted under USDOE sponsorship to obtain integral test data to validate the design methods used to predict fission product release from the core and plateout in the primary coolant circuit of a steam-cycle HTGR and liftoff during rapid depressurization transients. The depressurization test phase was initiated immediately after the 62-day irradiation phase. The tube side of one of the three parallel heat exchanger (HX) tube bundles was isolated to serve as an undisturbed reference for plateout predictions as well as to increase the maximum achievable shear ratio on the heat exchanger (by reducing the total flow area). Subsequently, the loop, including the remaining two HX bundles, were subjected to a series of blowdowns over a range of successively higher shear ratios.

The measured plateout distributions were compared to PADLOC code predictions (Medwid 1993). The comparison of the measured and predicted Ag-110m profiles was within the allowable uncertainty of 10x. However, Ag sorptivity at the highest temperatures (>~530 ^{o}C) was significantly overpredicted (the reference Ag isotherms are highly uncertain so this discrepancy is not surprising). Below ~530 ^{o}C, the Ag profile indicates perfect sink behavior, and the slope of the profile is well predicted (for perfect sink plateout, the deposition profile is a straight line on a semi-log plot, and the negative slope is inversely proportional to the mass transfer coefficient).

The measured Cs 137 axial profiles were flatter than those predicted by PADLOC, especially in the low temperature section (the Cs 134 profiles are essentially identical); PADLOC predicts more plateout at the heat exchanger hot end and significantly less plateout at the cold end. The comparison is within the allowable uncertainty of 10x, but the qualitative agreement is not good, especially for the T22 (2¼%Cr-1%Mo) section of the tube where the Cs profiles are almost flat, even showing apparent local maxima. There is also considerable tube-to-tube variation in the absolute activities; it is difficult to ascribe this variation to variations in the helium flow rates through the individual tubes because of the small variations in the specific Ag-110m activities from tube to tube.

In comparing the measured and predicted I-131 heat exchanger plateout distributions, the accuracy goal of 10x was not met. The slope of the axial plateout distributions differ, implying an incorrect temperature dependence for the sorption isotherm. Ironically, the sorption isotherm used in the PADLOC prediction was derived from iodine sorption measurements made at ORNL on T22 and at some of the lowest partial pressures attained experimentally to date. In principle, this iodine sorption isotherm should have been the most reliable of the isotherms used in this analysis.

The measured plateout profiles of Te-132 and Sb-125 exhibited perfect sink plateout behavior over the entire temperature range experienced in the BD-1 heat exchanger. The slopes of the profiles (i.e., the mass transfer coefficients) are well predicted by the PADLOC code.

Four *in-situ* blowdown tests were performed at the end of the irradiation at successively higher shear ratios of 0.72, 1.7, 2.8, and 5.6. In each blowdown test, the reentrained activity was trapped by a dedicated full-flow filter. The measured cumulative liftoff fractions were always <1% even at a shear ratio of 5.6. The liftoff fractions predicted with the POLO code (Bolin 1989) were much larger.

3.3 Materials Development and Qualification

The structural materials that may require further technology development for application in the NGNP are: (1) core graphites, (2) ceramics, (3) high-temperature metals and (4) reactor pressure vessel materials. The status of their development and qualification is summarized below. There is a vast literature on these topics, and the following discussion is meant only as a brief introduction.

3.3.1 Core Graphites

Graphite has been used as a moderator and a structural material for nuclear reactor cores since the dawn of the nuclear age. Certain graphite properties are of seminal importance to the proper functioning of the core. For example, several early graphite piles failed to go critical because of neutron-absorbing impurities in the graphite. The primary technical reason (there were political reasons as well) for closing the Hanford N-Reactor was the expansion and distortion of the core graphite. Stringent limits were upon primary coolant oxidants in FSV because concerns about oxidation of the PGX graphite core support floor which was aggravated by high iron impurities (Copinger 2004). Finally, disposition of irradiated graphite is a significant issue for the future D&D of graphite-moderated reactors at end-of-life (e.g., Neighbour 2002 and TCM-Manchester99 2001).

The graphite components of the reactor are the permanent side reflector, the core support structure and the core. The core includes the fuel elements and the replaceable reflector elements. For the commercial GT-MHR (Shenoy 1996), the reference material for the core, permanent side reflectors, and the core support is H-451 graphite which is no longer commercially available. The reference material for the permanent side reflector support blocks at the hot duct entrance and selected core support post blocks is a purified form of HLM grade graphite. Several billet sizes of these two graphite grades are required for different component sizes.

Both graphite grades of selected billet sizes were used successfully in the FSV reactor, and much is known about these materials. However, the NGNP design conditions are different from those in previous reactors such as FSV and the large HTGRs. Specifically for the NGNP, designing for conduction cooldown events requires graphite properties at higher temperatures, and the currently proposed structural design criteria require new material properties to be measured, and test data used to validate the design must satisfy more stringent QA requirements.

Due to changes in the graphite manufacturing industry, graphites made by different suppliers, and from new sources of raw materials must be characterized by testing before they can be qualified for use in the NGNP. Consequently, a high-priority DDN for the NGNP is to identify and qualify a near isotropic graphite equivalent to H-451.

The use of carbon/carbon (C/C) composites is proposed for several subcomponents in the control rod assembly. The selection was based on limited data from ORNL's work on irradiated C/C composite for fusion energy applications. C/C composite, therefore, needs to be further characterized by testing and its compatibility in the reactor environment needs to be assessed before it can be qualified for use in the NGNP.

The engineering development effort is therefore required in two major areas: (1) engineering properties to expand the database to cover NGNP specific considerations with the statistical significance required for the design, and (2) material supply to ensure a qualified source of raw materials and improve quality control.

The design of the graphite components is based on a considerable body of available graphite data. In the early 1970s, a near-isotropic, petroleum coke-based graphite, designated Grade H-451 and manufactured Great Lakes Carbon (now part of SGL Carbon), was developed for use in the HTGR. Numerous test programs and experiments were conducted, mostly by GA and ORNL, to characterize its behavior for commercial HTGRs. Consequently, a large, but incomplete, material property database exists for grade H-451. Work to obtain property data on HLM graphite has been relatively limited due to less severe service environment. All property data on H-451 and on HLM were documented by GA for the commercial HTGRs in the Graphite Design Data Manual (GDDM). These data will be used for the Conceptual Design (and perhaps Preliminary Design) of the NGNP core until a replacement graphite is characterized.

To supply the FSV reactor with graphite for reload fuel, a stockpile of coke was produced to make H-451. The coke was derived from a selected, high-purity petroleum feedstock, specially processed to provide a material with near-isotropic properties. Since the remaining FSV coke supply is insufficient for continued graphite production, SGL Carbon, the producer of H-451 graphite, has taken the initiative to perform screening tests on cokes from several new feedstocks and on graphite processed from these cokes. The purpose of these tests is to select cokes which will produce graphite with properties and microstructural characteristics that most closely match those of H-451 produced for FSV. SGL Carbon, in conjunction with ORNL, has selected three cokes for assessing the irradiation performance of the resulting graphites.

ORNL has evaluated various graphite non-destructive examination (NDE) techniques. For surface and subsurface flaw detection, these techniques include ultrasonic inspection, eddy-current techniques, and conventional and tomographic radiography. For graphite strength classification, they include sonic velocity, and attenuation measurements. Some data have been produced on the accuracy and limitations of these techniques for H-451 graphite. Although the existing database requires supplemental data to evaluate and validate, these techniques sufficient for use in production control of graphite. Eddy-current and ultrasonic measurements on H-451 graphite have significant potential as flaw detection techniques.

3.3.2 Ceramics

The graphite core support assembly includes hard ceramic pads that are located beneath the graphite elements (Shenoy 1996). The ceramic pads thermally insulate the underlying metallic core support floor from the hot helium gas in the core exit plenum. The reference material is a commercially available Aluminosilicate Ceramic, grade AD-85, manufactured by Coors Ceramics Company which is the same material used in the FSV reactor.

Initial FSV screening tests of a number of candidate ceramics, as well as data on the availability and cost for the sizes and quantities of materials required, led to the selection of alumina (Coors grade AD-85) blocks for the core support-pad insulation in the FSV reactor. From the design studies for the large HTGRs, the material selected for the upper and middle ceramic pads was grade AD-85 alumina. The bottom pad candidate material was polygranular fused silica. The alumina ceramics are stronger and are less subject to creep at high temperatures, while silica has the advantage of lower thermal conductivity at temperatures where its structural stability is adequate.

Mechanical and thermal properties of the FSV candidate hard ceramic materials were obtained in a helium environment. These include thermal conductivity, coefficient of thermal expansion, flexural strength, compressive strength, elastic modulus, Poisson's ratio, specific heat and resistance to thermally induced stresses.

3.3.3 High-Temperature Metals

Structural metals are used throughout the primary coolant circuits of HTGRs, including the reactor internals and heat exchangers. When the first HTGRs were designed, it was obvious that the metallic components would operate at high temperature and that some would be exposed to high neutron doses as well. The environmental aspect that was not fully anticipated was the and the extent to which the reactor primary coolant chemistry could vary.

In the earliest work in the 1960s and early 1970s, in support of the development of first generation HTGRs, the importance of environmental effects of primary coolant helium on and thermal aging of candidate HTGR alloys was recognized. Therefore, in addition to baseline mechanical property studies (primarily tensile strength, impact, and fatigue tests), high-temperature creep and thermal aging studies were performed on candidate HTGR alloys (Types 304 and 430 stainless steels, 2-1/4Cr-1Mo, Incoloy 800, and Hastelloy X) in impure helium (and in air for comparison) at temperatures up to 760 ^{o}C and for times approaching 18,000 hours. Since there were no operating HTGRs at that time, impure helium test environments were based on expectations. A typical impure helium test environment at that time contained 200 – 300 µatm of CO and H_2.

After the startup of the Peach Bottom, Dragon, and AVR reactors, it was learned that there could be significant quantities of other impurities present in helium primary coolants in addition to CO and H_2. In Peach Bottom, the primary coolant contained CH_4 (because of oil ingress from a leaking purified He transfer compressor), the Dragon primary coolant contained measurable H_20 and CH_4 (from steam generator leaks), and the AVR primary coolant exhibited considerable concentrations of CO_2 and CH_4. In view of the known carburization and/or oxidation (and decarburization) properties of CH_4 and/or CO_2, interest was renewed in the areas of primary coolant compatibility studies, and work began in the Europe using more realistic helium environments. This early work indicated significant environmental effects of simulated reactor helium on mechanical properties.

Partly because of the oil ingress experience at Peach Bottom, the FSV circulators were designed with water bearings rather than conventional oil bearings (a fateful mistake). These water bearings were the source of chronic water ingress events (Copinger 2004). These frequent water ingress events, and the excessive time required to recover from them, contributed substantially to a dismal lifetime capacity factor and to the early decommissioning of the FSV nuclear steam supply system (the plant was converted to natural gas and continues to operate). The use of water bearings and the attendant water ingress events produced a different coolant chemistry in FSV compared to Peach Bottom.

This history of unanticipated coolant chemistry in first-of-a-kind (FOAK) HTGRs has significance for the NGNP. With the elimination of the steam generator from the primary circuit and the use of magnetic bearings for the PCS and for the circulator, the potential for significant and persistent introduction of impurities into the primary helium appears to be quite low; consequently, the helium in the NGNP should have very low impurities. However, the oxidation potential (typically determined by the H_2/H_20 ratio in past HTGRs) is not easily predicted with high confidence. This uncertainty has considerable implications for conducting metals qualification programs and may need to be addressed by varying coolant chemistry over a wide range of oxidation potentials in the environment effects tests.

The history of metals performance in HTGRs is described briefly below; it is organized by reactor metals and heat-exchanger metals.

3.3.3.1 Reactor Metals

The major metallic reactor components include the metallic reactor internals and the hot duct assembly. All of these components are made of the high temperature metal Alloy 800H. Alloy 800H is an ASME Code-qualified material which was successfully utilized in the steam generators of the FSV plant.

Structural integrity of the Metallic Reactor Internals and Hot Duct components is required to support the reactor core and protect the primary pressure boundary from overheating during conduction cooldown events. Furthermore, premature removal and replacement of these components prior to the completion of the reactor design life would severely affect plant availability. The metallic components are exposed to neutron irradiation and to the temperatures and chemistry of the primary coolant during the life of the plant. Sufficient test data are needed to quantify these effects on the mechanical properties of Alloy 800H base metal and weldments so that the appropriate margin can be included in the design. This database will be coupled with an in-service surveillance program and a design that assures component replaceability.

The design of the reactor metal components is based on the ASME Code for Alloy 800H with conservative reductions in Code allowables based on existing data relative to environmental effects on this alloy. Since the early 1960s, numerous test programs and experiments have been conducted in support of metals technology for the large HTGR, primarily in the United States, United Kingdom, Federal Republic of Germany, France, and Japan. These programs have included extensive testing of many metallic materials, including Alloy 800H, to determine and to quantify the effects of thermal and primary helium exposure on mechanical behavior, and, to a much lesser extent, the effects of neutron irradiation. Through these efforts, it has been observed that long-term exposure in the thermal, neutron, and primary coolant helium environment of a gas-cooled reactor can degrade the mechanical properties of metallic materials.

3.3.3.2 Heat Exchanger Metals

The principal heat exchangers for the NGNP are the shutdown cooling heat exchanger (SHE) , the IHX, the recuperator, and the precooler and intercooler in the PCS, and the process heat exchangers in the hydrogen plants. The status of some leading candidate materials is briefly summarized below.

The key metallic components of the SHE are currently specified as 2¼Cr-1Mo steel. The material 2¼Cr-1Mo is ASME Code qualified for use in safety-class (Section III, Class 1) pressure boundaries at the expected SHE service temperatures. However, the shrouds for the SHE will experience very high temperatures during normal reactor shutdown and conduction cooldown conditions and may be required to be fabricated from a material with higher temperature capabilities.

A leading candidate material for the recuperator is Type 316L stainless steel, a low carbon, austenitic stainless steel selected for its excellent fabricability (brazability) and adequate strength at the operating temperatures.

A leading candidate material for the precooler and intercooler is ½Cr-½Mo steel, selected for its adequate strength for this low-temperature application and its excellent fabricability (weldability). The material ½Cr-½Mo is ASME Code qualified for use in safety-class (Section III, Class 1) pressure boundaries at the expected service temperatures. The effect of helium or water on precooler and intercooler materials is not expected to be a concern because of the low operating temperatures.

Extensive laboratory testing, using a range of temperatures and helium impurity levels (H_2, H_20, C, $C0_2$, and CH_4), has been carried out in the USA, Europe, and Japan over the past three decades to verify the performance of a variety of high-temperature materials in the primary helium environments for HTGR systems. Test materials included wrought alloys such as 2¼Cr-1Mo steel, Alloy 800H, Hastelloy X, Inconel 617 and other metals. The basic conclusion resulting from this work was that, for MHR primary coolant conditions at temperatures up to approximately 538 ^{o}C for 2¼Cr-1Mo and 704 ^{o}C for Alloy 800H (and also Hastelloy X and Inconel 617), no significant environmental degradation effects requiring modifications to design allowables were likely over the life of the plant. However, confirmation of the resistance of SHE and recuperator materials operating within and above this range of temperatures, for the specific impurities in the NGNP primary coolant, will be needed.

Limited laboratory testing has also been carried out to determine the effects of various fission products (Cs, Te, I, etc.) released from the core on the behavior of structural alloys at elevated temperatures. The results of these studies indicate that fission product/metal interactions can occur in reactor environments and that these interactions can, in certain cases, adversely affect the mechanical properties of metallic reactor components at sufficiently high concentrations. The degree of fission product attack and its subsequent effect on alloy properties is a function of many variables including fission product species and concentration, alloy type and structure, exposure time, temperature, and presence of other species (e.g., carbon, oxygen, other fission products, etc.). The results of these studies have shown that I, Cs, and Te can cause embrittlement of stainless steels and some nickel-base alloys at temperatures as low as 400 to 450 ^{o}C. These fission product corrosion results are primarily from the previous molten-salt reactor program at ORNL wherein the concentrations of fission products were orders of magnitude higher than anticipated in the primary circuit of the NGNP. Although still subject to experimental confirmation, radionuclide corrosion of the IHX or the turbine blades is not expected to be a significant problem for the NGNP.

Corrosion testing has also been conducted to determine the effects of other impurity elements which may be released from the fuel and graphite (e.g., Cl and S[9]) on the behavior of structural

[9] Trace quantities of H_2S were introduced into the FSV primary coolant by the hydrolysis of metal sulfide impurities in the petroleum pitch-based fuel compact matrix during water ingress events. The possibility

materials. Tests in helium containing small concentrations of H_2 and H_2S have indicated that the temperature threshold for sulfidation attack of many structural materials (304 stainless steel, 2¼Cr-1Mo steel, Alloy 800, Inconels 600 and 625) under these conditions is about 510 ^oC, and that liquid phase formation resulting from attack by sulfur species (H_2S) can occur on nickel-containing alloys at temperatures as low as 638 ^oC. Tests on carbon steel, stainless steel, and nickel-base alloys in dry chlorine-containing environments (HC1) indicate that significant corrosion of these materials can occur at temperatures of approximately 260, 340, and 425 ^oC, respectively, via the formation of metal chlorides, some of which (e.g., $MnCl_2$) can melt at temperatures as low as 650 ^oC. Chlorine in aqueous solutions, via reaction between chlorine and moisture, can also result in stress corrosion cracking of many metallic materials and has been observed in stainless steel components in FSV.

Extensive programs have been carried out to assess the effects of cold work, short-term high-temperature annealing (at 1010 ^oC) and recrystallization on the strength properties of Alloy 800H. The results of these investigations showed that low creep ductility and high residual stress can be induced in the material by cold work, and that high-temperature annealing and recrystallization can restore the properties of Alloy 800H to that of pre-cold-worked material. Confirmation of such effects for the material selected for the uncooled components of the SHE should be obtained.

3.3.4 Reactor Pressure Vessel Materials

The NGNP vessel system consists of the reactor pressure vessel (RPV), power conversion vessel (PCV), intermediate heat exchanger vessel (IHXV) and two cross vessels (XVs). These vessels are part of the primary coolant boundary and would be classified as safety related.

The NGNP reactor pressure vessel is planned to be fabricated from 2¼Cr–1Mo steel because it is expected to experience temperatures that exceed the capability of steels commonly used for LWR reactor vessels (e.g., SA533 Grade B Class 1 plate and SA508 Class 3 forgings of manganese-molybdenum steel). Likewise, the IHX vessel and cross vessels are also planned to be fabricated from 2¼Cr–1Mo steel because they are expected to experience temperatures similar to those in the reactor vessel. However, the PCS vessel temperatures are sufficiently low that it can be fabricated from the LWR vessel steel. Bolting material will be SA540 Grade B24 Class 3 nickel-chromium-molybdenum steel. All of the selected materials are commonly used for pressure vessels. 2¼Cr–1Mo steel is a code-qualified RPV material.

Alloy 9Cr-IMo-V ferritic steel (SA-387 Grade 91, Class 2 for plates and SA-336 Grade F91 for forgings) is the recommended backup to 2¼Cr–1Mo steel for the NGNP RPV and a potential product improvement for follow-on commercial H2-MHRs.

The 9Cr-IMo-V alloy was developed by ORNL and Combustion Engineering (CE) during the 1970s and 1980s to extend and optimize its high-temperature properties. Although the material

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of sulfur impurities in the NGNP primary helium has been much reduced by the adoption of high purity, phenolic resin-based matrix for the fuel compacts.

has been primarily used as tubing and piping in fossil energy applications, the developers of the steel initially envisioned its use in nuclear steam generators and vessels for liquid-metal reactor applications. Later, it was contemplated for use in fusion energy systems. The use of 9Cr-IMo-V has also been proposed by the French for the construction of heavy-wall vessels for refinery applications, and thicknesses up to 300 mm have been produced.

Significant data exist characterizing the effects of fast neutron (E > 0.1 Mev) irradiation on the nil-ductility transition temperature (NDTT) and other mechanical properties of 9Cr-1Mo-V. Samples have been irradiated in LMFBR and fusion reactor irradiation environments (EBR-II, FFTF, and HFIR), and data have been obtained on tensile, creep, and fracture toughness properties, as well as on changes in microstructure and swelling. These data indicate that 9Cr-IMo-V will perform as well as LWR pressure vessel steels with respect to irradiation properties.

An extensive database has been developed by ORNL and CE for the high-temperature mechanical properties of 9Cr-IMo-V including tensile, creep, fatigue, toughness (and corrosion) measurements on as-received, heat-treated, and thermally-aged base metal and welds. However, these measurements have been concentrated on plate materials (and weldments) of relatively small thickness (200 mm) compared to the expected reactor pressure vessel thickness (300 mm).

3.4 Heat Transfer Technology

Historically, the heat transfer technology relied upon in gas-cooled reactors has typically been based on helical-coil heat exchanger technology. Helical-coil heat exchangers are the preferred design choice for steam generators and are also used in emergency core cooling system, auxiliary cooling systems, and shutdown cooling systems. Helical-coil heat exchangers often have a relative large logarithmic mean temperature difference (LMTD) in order to reduce the heat transfer surface area and size of the heat exchanger.

Printed circuit heat exchanger (PCHE) technology achieves high effectiveness and low LMTD in a compact heat exchanger with reasonable pressure drops across the heat exchanger. PCHEs consist of alternating metallic plates in which microchannels have been chemically etched and then joined together under high pressure and temperatures to form a diffusion-bonded heat transfer core. PCHE technology has been applied to numerous industries but has yet to be applied in the nuclear industry – especially for gas-cooled reactors at the very high temperatures.

As part of JAEA's plan to demonstrate hydrogen production with the High Temperature Engineering Test Reactor (HTTR), a high-temperature isolation valve (HTIV) must be installed in the secondary helium hot gas duct, which penetrates the reactor containment vessel. Development of the HTIV is underway. The technical issues are as follows: (1) prevention of the valve seating from thermal deformation, (2) development of a new material for the valve seat surface, and (3) selection of a valve seat structure having a high sealing performance. An angle valve with an inner thermal insulator was selected. A new valve seat material, with sufficient hardness and wear resistance over 900 ^{o}C, was developed based on the Stellite alloy that is used for valves at around 500 ^{o}C. A component test of the valve seat indicates that a flat type valve seat can maintain the face roughness of the valve seat within allowable limits during operation. A 1/2 scale model of HTIV was fabricated to confirm seal performance and structural integrity. The He leak rate was confirmed to be less than the target value (Nishihara 2004).

3.5 Power Conversion System

Early versions of the MHR utilized a power conversion system based on the Rankine cycle (i.e., steam cycle), but a direct Brayton cycle was adopted as part of the design evolution that was driven by the need to make the MHR more economically competitive with other electricity generation options. The initial preconceptual GT-MHR design was developed under a joint initiative of the DOE and US Utilities over the period 1991 - 1994. A vertical integrated power conversion system design was selected from trade studies performed as part of the GT-MHR preconceptual design development.

In 1994, the GT-MHR was selected as the basis for a joint effort by the U.S. and Russia to design an MHR for disposition of surplus weapons-grade plutonium (that would be used as the fuel for the reactor). The Experimental Design Bureau for Mechanical Engineering (OKBM) in Nizhny Novgorod was given responsibility for GT-MHR design development and is the Chief Designer of the reactor plant. In support of this arrangement, DOE has also negotiated a contract with OKBM to perform R&D work (see Section 6.4 for a description of this program).

Helium is an ideal working fluid for a nuclear gas turbine because it does not become radioactive and has excellent heat transfer properties. A substantial database exists regarding an understanding of how the unique properties of helium will be addressed in the design and operation of the rotating machinery. While the properties of helium influence the gas flow path geometries, the aerodynamic and structural design procedures used are similar to conventional air-breathing aeroengine gas turbine practice. The high specific power associated with helium operation, together with the high gas pressure in the closed helium loop, results in a machine size that is physically smaller than industrial and aeroderivative gas turbines currently in utility service. The design and operation of high temperature systems with helium as the working fluid are well understood.

3.5.1 Major PCS Components

The major components in the power conversion system are based on combustion gas turbines (both industrial and aeroderivative units) that are in service today for electrical power generation. The major components include the following: (1) turbocompressor, (2) magnetic bearings, (3) electrical generator, (4) recuperator, (5) precooler/intercooler, and (6) pressure vessel. The PCS uniquely packages together the major components to achieve a highly efficient power conversion unit. The fact that the major components are based on proven hardware reduces development risk for the power conversion system (e.g., McDonald 1994a, McDonald 1994b, McDonald 1995, etc.)

3.5.1.1 Turbomachine

The turbomachine consists of a helium turbocompressor and generator housed in the Power Conversion Vessel (PCV). The turbocompressor consists of a two-section compressor (separated to facilitate intercooling) and turbine. The turbocompressor drives a submerged generator. The vertical turbomachine rotor assembly is supported on an active magnetic bearing system (four radial journal bearings and a thrust bearing). The bearing system also embodies catcher bearings capable of accommodating a number of rotor drops, with no damage to the rotor, in the event of loss of the magnetic field.

Significant turbomachinery technology is available from the gas turbine industry, both for industrial and aeroderivative units. Typical examples of existing machines are two General Electric gas turbines, (1) the MS9001F industrial gas turbine with a rating of 226 MW(e) and (2) the LM6000 aeroderivative gas turbine derived from the CF6-80C2 commercial aircraft engine rated at 42 Mw(e). State-of-the-art technology from these engines is directly applicable to the design of the helium turbomachine, particularly in the areas of design methodology, performance, materials, and fabrication methods.

Turbomachine component and assembly test data for the NGNP design are needed to verify that mechanical, electrical, and electronic equipment meet overall plant performance and reliability requirements for the service conditions anticipated.

3.5.1.2 Magnetic Bearings

In recent years there has been substantial use of magnetic bearings in industrial applications worldwide and active magnetic bearing technology is already well proven in sizes similar to those employed in the NGNP design concept. They have already established an excellent track record in the gas pipeline and petrochemical industries, contributing greatly to reducing unscheduled shutdowns, process flow contamination, and maintenance costs. Today, several million hours of operating time has been accumulated on active magnetic bearings. Over 150 large turbomachines (e.g., gas compressors, gas turbines, turboexpanders) have run millions of hours; the technology is regarded to be mature enough for the NGNP turbomachine.

3.5.1.3 Electrical Generator

Electrical generators with ratings of over 200 MVA are in service today for 60 Hz gas turbines. The only major differences in the GT-MHR application are the vertical rotor assembly and the use of helium for cooling of the stator and gap. Modifications to the structural support for the stator windings will accommodate vertical installation. Submerged electric motor drives have operated successfully in a helium environment (e.g., AVR and THTR plant circulators), and only the specific vendor’s insulation system needs confirmation.

3.5.1.4 Recuperator

The PCS recuperator is a compact, high effectiveness, gas-to-gas heat exchanger. Recuperators are used extensively in applications with propulsion and pipeline gas-turbine engines. In these applications (especially with propulsion engines), the recuperators are subjected to many more start/stop and low-load/high-load cycles than will be seen in the NGNP application, and the propulsion system transients are much more severe. Much of the experience data from these applications, therefore, can be directly used to evaluate the NGNP recuperator design. It is of interest to note that a plate-fin recuperator was used in the US Army 400 KW(e) ML-1 plant, the only nuclear closed-cycle gas turbine to have operated.

Recuperator performance test data for the NGNP design are needed to verify that mechanical and heat transfer characteristics meet overall plant performance and reliability requirements for the service conditions anticipated.

3.5.1.5 Precooler/Intercooler

The PCS precooler and intercooler are helium-to-water once through helical coil heat exchangers. Helium flows on the outside of the tubes counterflow to the water on the inside of the tubes. The precooler is located at the inlet of the first stage compressor, and the intercooler is located at the inlet of the second stage compressor.

The technology for helium-to-water tubular heat exchangers is well established from FSV and THTR as well as from fabrication development programs performed for the NP-MHTGR. The development program for the NP-MHTGR had also established a reference Tube Restraint Device (TRD) design for helical coil tube bundles which was extensively tested. Assembly methods and procedures for the TRD were also developed. This design should, to a large extent, be applicable to helical coils with fins provided the fins are low profile. Other development programs and tests performed for the NP-MHTGR applicable to the precooler/intercooler include steam generator flow distribution tests, flow induced vibration tests, helical tube welding methods and tube inspection (ISI) methods development.

Precooler and intercooler performance test data for the NGNP design are needed to verify that mechanical and heat transfer characteristics meet overall plant performance and reliability requirements for the service conditions anticipated.

3.5.1.6 Pressure Vessel

Conventional SA508/533 ferritic steels that have been extensively used for light water reactor RPVs can be used for the PCS pressure vessel since there is cold high-pressure helium available to cool the vessel during normal operation and the PCS PV is effectively isolated from the reactor core during pressurized- and depressurized core heatup transients. Moreover, the PCS PV receives a negligible fast neutron dose since it is in a separate cavity.

The ASME Code rules for SA508/533 are already established, and no further R&D appears to be required for their use as the PCS PV material. The Code limits the application of this material to approximately 370 ºC for a Class 1 boundary, for an unlimited time, with very limited provisions made for shorter time exposure beyond this temperature.

3.5.2 He Turbine Operating Experience

In the 1970s, two helium facilities were built and operated in Germany; the roles that they played in the European studies of a nuclear gas turbine plant in that era have been discussed previously (Noack 1975, Weisbrodt 1995).

A high temperature test facility (HHV) was operated at KFA, Juelich, FRG, in which a section of a helium gas turbine (300 MW_e size) was tested with a turbine inlet temperature of 850 °C with the capability to go to 1000 °C. The turbine and compressor had two and eight stages, respectively. For operation at 850 °C, uncooled turbine blades of Nimocast 713LC were used, these being formed by a precision casting procedure. A complex internal cooling system was used to keep the discs and blade root attachments to acceptable temperatures commensurate with stress limitations. The turbine power was only 45 MW, and since the compressor needed 90 MW, an additional 45 MW was provided by an electric motor. It also included a loop for testing the hot gas duct insulation. Helium quality in the test loop corresponded to reactor conditions, except for the radioactive contamination. This test program confirmed the compressor and turbine aerodynamic performance, rotor structural integrity, and bearing performance.

The Oberhausen II 50 MW(e) helium turbine plant had a coke oven gas-fired heater and provided electrical power and district heating to the city of Oberhausen. Metallurgical/stress limitations in the externally gas-fired heater limited the turbine inlet temperature to 750 °C. The turbine had seven high-pressure (HP) and 11 low-pressure (LP) stages. The plant's hot end components were designed for 100,000 hour operation at 750 °C. In the high temperature turbine, the blade material in the first two stages was UDIMET 520, and the remaining stages were NIMONIC 90.

After 20,000 hours of operation, a rotor blade in the first stage of the HP compressor failed at the root. In passing through the compressor, it did extensive damage necessitating replacement of the complete compressor rotor. The Oberhausen II helium turbine plant was plagued with other problems as well and produced only 30 MW instead of the 50 MW design value (Weisbrodt 1995). When the supply of coke-oven gas from a nearby steel mill was no longer available, the plant was decommissioned and dismantled.

3.6 Design Verification and Support

As described in Section 2.3.3, the base technology for designing most MHR systems, structures and components derives from five decades of international R&D programs combined with the design, construction and operation of seven He-cooled reactors. For the NGNP preconceptual design, the exceptions are the PCS, IHX and hydrogen plants which are discussed separately. Nevertheless, there are design-specific features of some SSCs that will require design verification by testing with semi-scale mockups or with actual prototypical components. The DV&S DDNs are described in Section 5.

The status of DV&S testing for important NGNP SSCs is divided into six subsections, each corresponding to a major plant SSC as follows:

1. Fuel Handling System (FHS)
2. Neutron Control System (NCS)
3. Shutdown Cooling System (SCS) Shutdown Heat Exchanger (SHE)

4. Shutdown Cooling System Circulator
5. Reactor Cavity Cooing System (RCCS)
6. Reactor Internals and Hot Duct.

3.6.1 Fuel Handling System

The Fuel Handling System (FHS) is an automated set of computer controlled machines, which refuel the reactor module.

A large experience base exists from designing, building, testing and operating fuel handling equipment for the Peach Bottom and FSV reactors. Although the Peach Bottom fuel handling machine was manually operated, important technology was developed in the areas of: (1) electrical power and signal cables for operation in 230 °C helium with high gamma background; (2) lubricants for use in the same harsh environment; (3) electronic sensors for use on the grapple head; (4) grapple head floating plate technology for light touch in horizontal and vertical directions; and (5) general purpose manipulator technology adapted for special use in the reactor.

The FSV Fuel Handling Machine (FHM) was designed and built in the late 1960s during the time that programmed machine tools were being developed for numerical control. This machine advanced from the Peach Bottom 1 technology in areas of: (1) computer control of multiple positioning systems in automatic mode or direct operator control in manual operation mode; (2) the use of electric motors, brakes, and position feedback instrumentation in a helium environment; (3) the use of a radiation-hardened television camera and lighting in helium; (4) programming techniques to safely operate the FHM within limits set by hard-wired interlocks and, (5) elementary inventory control, which was greatly enhanced in a 1989 control system upgrade.

The fuel handling equipment concepts for the NGNP have evolved from this technology. The fuel handling machine has motions similar to those of FSV along with several new automated machines that operate in concert. Simultaneous operations of several machines are planned to refuel the reactor within the allocated time objectives.

The Fuel Transfer Cask (FTC) and the Element Hoist and Grapple Assembly robot are all new design concepts required to operate in a helium environment. These machines incorporate proven technology where applicable. For instance, the FTC will use the grapple head, telescopic guide tubes and isolation valve designs similar to those used in the FSV FHM.

The Fuel Handling Equipment Positioner (FHEP) is similar to a commercially available, computer operated gantry crane with position control of the x, y, z and load rotation axes. The FHEP will automatically transport, position, couple and uncouple the portable fuel and target handling equipment.

The Element Hoist and Grapple Assembly robot and its end effectors are similar to the gantry robots applied by GA in the US Army chemical weapons demilitarization development program. General Atomics has developed the robotics for the remote handling of munitions in a lethal agent environment. The particular relevant expertise gained and "lessons learned" in the design, use and control of multiple gantry robots, end-effectors, and decontamination compatible hardware is available and applicable to the gantry robots to be used in the Local Refueling and Storage Facilities and Fuel Sealing and Inspection Facility.

The computer control and element accountability system will utilize background data derived from the FSV project, commercial HTGR designs, the GA Demil program and industrial applications of computer controlled equipment. The FSV and Demil projects provide tested databases for the NGNP computer architecture which include automated serialized accounting of fuel elements and target assemblies. In 1989, the FSV computer hardware and software were expanded and improved with particular attention to fuel element accountability. The ability to read serialized characters on the fuel elements, construct and maintain real time accounting files by location and item identification utilizes character recognition and real-time database information.

Data from testing of FHS components specifically designed for NGNP are needed to verify that mechanical, electrical, and electronic equipment meet overall plant performance and reliability requirements for the service conditions anticipated.

3.6.2 Neutron Control System

The Neutron Control System (NCS) controls reactivity within the reactor and must shut the reactor down upon command with high reliability. The NCS consists of control rod drives, neutron detection equipment, reserve shutdown control equipment, and electronic control system.

The design of the NCS for the NGNP is based on previous designs developed for FSV, the large commercial HTGR, the MHTGR and the NP-MHTGR. An extensive R&D program was carried out prior to final fabrication and installation of the FSV NCS.

Engineering development is necessary for the NGNP because physical configuration and performance requirements are different from the design used at FSV. As designed, the NGNP reactor core is taller and the stroke of the control rods is greater. Interfaces between the NCS components and reactor vessel are different as are operating temperatures and pressures. Data from the previous designs provide the foundation for the NGNP design, but the design will include new technical developments, such as brushless DC-motor controllers, that need to be verified.

Data from testing of NCS components specifically designed for NGNP are needed to verify that mechanical, electrical, and electronic equipment meet overall plant performance and reliability requirements for the service conditions anticipated.

3.6.3 Shutdown Cooling System Heat Exchanger

The Shutdown Cooling System (SCS) shutdown heat exchanger (SHE) provides decay heat removal when the power conversion system is unavailable.

The NGNP SHE preconceptual design is similar to the FSV steam generator. The heat transfer testing and operational data for the FSV steam generators have confirmed that the heat transfer correlation used for the shell side heat transfer slightly under predicts the shell side heat transfer coefficient. Thus, it should not be necessary to perform additional heat transfer tests for the SHE.

During the NP-MHTGR program, design requirements for the SHE were established, and the SHE arrangement and basic features were defined. The focus of attention was on interfaces with the reactor vessel and reactor building. The basic need for the NGNP SHE testing is to verify the design satisfies performance requirements under all operating conditions.

3.6.4 Shutdown Cooling System Circulator

The SCS circulator consists of a submerged radial flow compressor with integral electric motor. Speed control is provided by an external variable frequency controller. The design and development of the circulator is based on that done for the AGR plant circulators in the U.K.

In support of the AGR plants in the U.K., Howden has fabricated 116 circulators, which have accumulated over 7.5 million hours of operation. In recent years these machines have demonstrated an availability of >99 % (i.e., less than 1% of the plant downtime was attributable to the circulators). Applicable experience gained from the AGR circulators can be utilized in the design of the NGNP SCS circulator.

As part of the NP-MHTGR circulator program, a new impeller geometry (designated L3R) was designed by Howden and tested to confirm that it would meet design requirements. To facilitate the testing in an existing facility, it was necessary to fabricate and test a smaller size impeller (i.e., 1 meter diameter) than the actual machine size. The tests confirmed the performance of the basic impeller geometry. It is anticipated that the same geometry can be utilized in the NGNP shutdown cooling circulator.

Significant development work and testing were performed on the circulator designs for the NP-MHTGR. Many subcomponent tests for the main circulator were completed, and the following data are directly applicable to the NGNP circulator: (1) performance data from model impeller (1 meter diameter) and (2) electric motor insulation performance (integrity confirmation) in a helium environment. Verification of the design is needed to demonstrate the NGNP SCS circulator meets performance requirements.

3.6.5 Reactor Cavity Cooling System

The Reactor Cavity Cooling System (RCCS) transports the core residual and decay heat from cooling panels in the reactor cavity to the environment during conduction cooldown events. During normal operation, the RCCS maintains temperatures in the reactor cavity and surrounding concrete at acceptable conditions.

Thermal hydraulic analyses of passive RCCSs similar to that employed in the NGNP preconceptual design have been performed through computer modeling of the heat and mass transport phenomena that govern the system operation. Several of the key physical processes and components that support the passive nature of the system need to be tested to validate the adequacy of these computer models, and to confirm system performance.

3.6.6 Reactor Internals and Hot Duct

The reactor internals and hot duct components include the graphite core and core supports, control rods and control rod shock absorber, and hot duct and fibrous insulation. These components must withstand significant mechanical loads during normal operation and design basis accidents.

Significant development and testing of reactor internal components to verify the HTGR design has been completed in prior years. A series of dynamic tests of unirradiated HTGR hexagonal fuel elements was performed for the 3000 MW(t) HTGR with a pendulum rig. Those tests, mostly on H-327 graphite blocks, indicated that the dynamic strength of a fuel element can be predicted reasonably well with static finite element methods, for a maximum relative impact velocity of 3 m/s or less.

The failure loads and failure modes of unirradiated HTGR fuel elements under mechanical loading were determined for FSV elements and for large HTGR elements. Some analytical correlations were performed, but they did not include crack progression analyses. Limited cracking under thermal and irradiation stresses was observed for two FSV fuel elements (Baxter 1994), and reasonably good analytical correlation has been achieved. The cracking was far from being sufficient enough to represent failure in a functional sense. NRC review of these cracked FSV fuel elements confirmed this conclusion.

Extensive core seismic tests on 1/5- and 1/2-scale core models, including a 1/5-scale full array model subjected to time history excitations, were performed for the large HTGR. Data obtained were core assembly dynamic characteristics and loads, including resonance behavior, core deflections, core support and core element loads, and fuel element dowel loads.

Full-scale testing of FSV core support structure models was performed to confirm adequate safety factors for vertical loads. A series of tests on large HTGR core support posts and seats did not correlate well with analytical predictions. Initially, premature failure of the seats was experienced. After redesign of the seats, the experimental ultimate load exceeded the analytical predictions. Subsequently, a detailed three-dimensional analysis improved these correlations.

Tests on the control rod shock absorber used for the FSV reactor was performed. The shock absorber was a specially designed metallic, bellows-shaped unit, which was attached to the bottom of the control rod. Testing was performed in ambient air conditions by dropping the control rod onto a graphite support member. The shock absorber absorbed the energy without failing the graphite core support. Testing was limited to unirradiated metallic bellows.

Flow induced vibration testing of a 0.45-scale model representing three regions and several reflector columns of the FSV core was performed. In this test, the column elements were graphite, and air was the working fluid. Flow- induced vibrations were encountered under some test conditions. Under the NP-MHTGR program a 1/4-scale model of a single core column made of plastic was also tested in air. Flow-induced vibrations were encountered at extremely high flow rates and most often the column vibrated without making contact with the surrounding flow channel. In both of these tests, vibrations were eliminated completely by constraining the topmost column element. The NGNP core design incorporates similar constraints at the top of each core column. However, the core is taller and flow conditions are different requiring additional flow testing to confirm that performance requirements are met.

A series of tests on fibrous insulation were performed for the FSV reactor to evaluate candidate materials, as well as to characterize the final materials chosen for the reactor. A number of different quartz and alumina silicate fibrous insulation materials were tested. The material capabilities addressed in these tests included material handling, chemical analysis, resiliency and relaxation, water effects, steam effects, neutron irradiation effects, thermal conductivity, flow-induced erosion, helium flow permeation, and fatigue caused by acoustic and mechanical vibrations.

After commissioning the FSV reactor, the fibrous insulation test program continued until the mid-1980s but was directed at various large HTGR designs. These tests incorporated new, more advanced materials as they became available. Some of these fibrous insulation materials demonstrated superior resiliency and relaxation characteristics and a higher tolerance to vibration. However, the processing methods for several insulations have changed since these tests were performed, such that the properties of the currently available materials are not known accurately. These properties need to be determined to confirm the material properties used in the design of the NGNP.

Previous extensive tests from 1968 to 1985 on fibrous insulation blankets sandwiched between steel plates for protection of the pressurized reactor vessel were performed. The insulation was held in place by steel coverplates and seal sheets on the hot-side surface. These tests produced useful design data, but the requirements, dimensions and construction for the NGNP are different and limited component testing of the design is required.

Component test data are needed for NGNP specific component designs and design conditions to validate analytical performance predictions and confirm that the components will function without failure for the duration of their service lives.

3.7 Hydrogen Production

3.7.1 Sulfur-Iodine Thermochemical Water-Splitting

The US DOE research and development effort on the sulfur-iodine cycle has been done primarily in collaboration with the French Commissariat à l'Energie Atomique (CEA) under an International Nuclear Energy Research Initiative (I-NERI) agreement since 2003. There is close coordination between the project participants in developing the three component reaction sections: the H_2SO_4 decomposition section, done by Sandia National Laboratory; the HI decomposition section, done by GA; and the Bunsen reaction equipment, provided by CEA. Each participant has designed and constructed their respective section and is working to integrate them in a SI Integrated Laboratory-Scale (ILS) experiment. This experiment is on track to begin integrated operations in late 2007. All experimental equipment has been transported to the GA site and is currently undergoing assembly and integration. Through 2004 and 2005, experimental work in glass equipment was conducted to evaluate and choose appropriate methods for carrying out the reactions in each section. Design work in 2006 allowed for lab-scale devices to be constructed in 2007 from engineering materials that are expected to be used in a pilot-scale hydrogen production facility scheduled for operation beginning in 2013. These lab-scale devices make up the equipment of the ILS experiment. Unlike previous demonstrations elsewhere, the ILS experiment will operate at temperatures and pressures expected to be seen at larger scales. The ILS test is expected to operate at least through the end of 2008.

The highly corrosive nature of the chemical streams in the SI process has led to significant research work in the area of materials compatibility. Early screenings showed that alloys of tantalum appeared suitable, and current work is exploring long-term performance and corrosion resistance of materials stressed or machined in ways that materials of construction for larger scale plants will experience. Devices for testing materials under simplified flow conditions have been built. The ILS experiment will also be a test bed for corrosion resistance of engineering materials during its operation.

Modeling and simulation of the SI process is necessary to predict thermodynamic efficiency, and to size equipment for cost estimation. Any uncertainties in the model are retained in efficiency and cost calculations. Work at Clemson University funded under a NERI grant is collecting thermodynamic data that will continue to improve the robustness of modeling and simulation efforts.

High-temperature inorganic membranes are being developed for use in the separation of SO_2 and O_2 from other chemical species in the high-temperature decomposition of H_2SO_4. This separation has the potential to shift the equilibrium of the reaction resulting in a potentially lower reaction temperature or increased process efficiency. The use of membranes for dewatering process streams is also being investigated. Most importantly, the removal of water from a mixture of water, elemental iodine, and hydriodic acid (HI) is being studied.

Catalysts are also being developed that will be highly active and stable in the harsh acidic environments and high temperatures encountered in the SI process. Iron oxide catalysts for sulfuric acid decomposition are suitable at higher temperatures (above 870 ^{o}C), and platinum-based catalysts can be used when the peak process temperature is below 870 ^{o}C. Platinum-based catalysts are not suitable for use in HI decomposition reactors, however activated carbon catalysts have been shown to be effective and inexpensive.

3.7.2 High Temperature Electrolysis

Solid-oxide electrolyzer (SOE) concepts based on both planar-cell and tubular-cell technologies are currently being developed. SOE technology based on the planar-cell concept is being developed as part of the Nuclear Hydrogen Initiative and involves collaboration between INL and Ceramatec of Salt Lake City, UT. A potential issue for the planar-cell concept is stack durability and sealing as the result of thermal cycles. Tubular cells have less active cell area per unit volume than planar cells but are less susceptible to this issue. Toshiba Corporation is currently developing an SOE concept based on tubular-cell technology. The GA team believes both the planar-cell and tubular-cell technologies are promising concepts for future commercialization and recommends that both concepts be developed through at least the pilot-scale demonstration stage so that tradeoffs between capital costs and long-term performance can be accurately characterized. The technology development status for both concepts is summarized below.

3.7.2.1 Tubular-Cell Technology

Figure 3-4 shows a prototype tubular cell manufactured by Toshiba. The electrolyte is Yttria-Stabilized Zirconia, the anode (oxygen electrode) is LSM (Strontium-doped Lanthanum Manganite), and the cathode (hydrogen electrode) is Ni-YSZ (a mixture of metallic Nickel and Yttria-Stabilized Zirconia). Toshiba has completed preliminary testing at 800 °C of an SOE unit consisting of three blocks containing five cells each (total of 15 cells). Test results showed nearly uniform open-cell voltage (OCV) and area-specific resistance (ASR) among the three blocks. The unit produced 130 normal liters per hour of hydrogen, which was above its design value of 100 liters per hour. Toshiba has also developed a three-dimensional CFD model of the unit (using the STAR-CD® code) and has obtained good agreement between model results and test data.

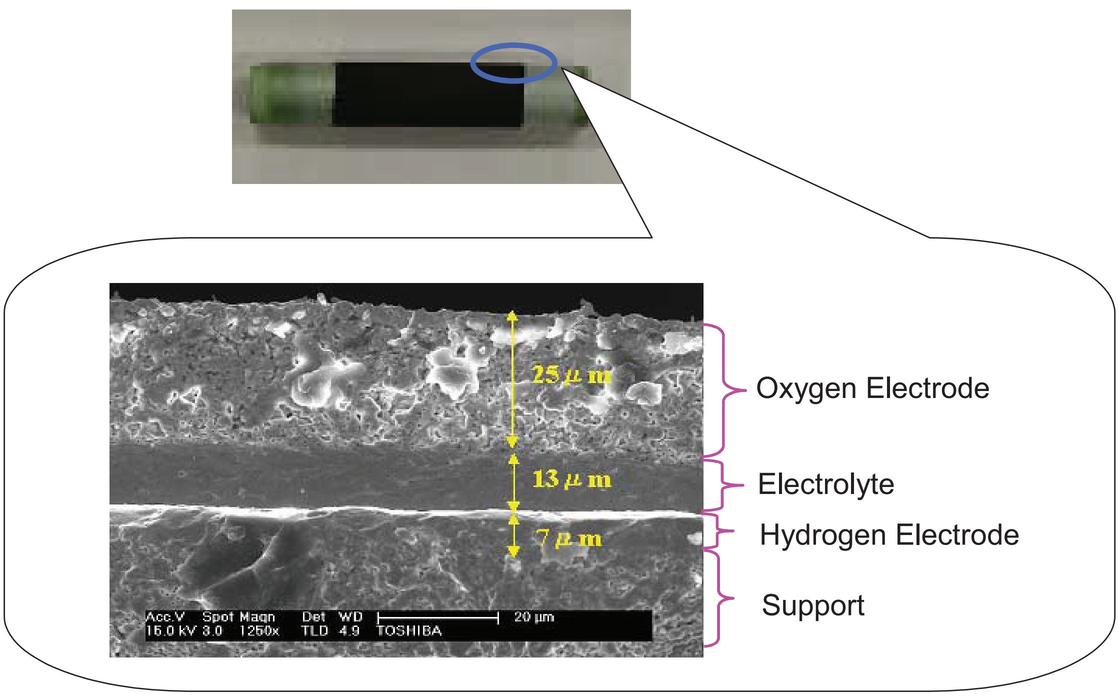
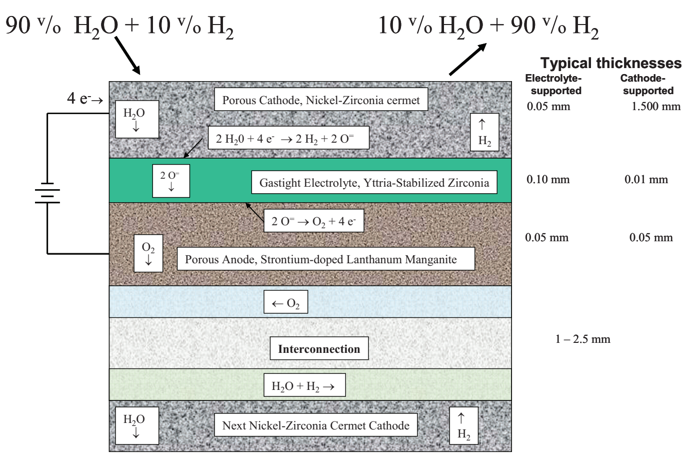


Figure 3-4. Prototype HTE Tubular Cell Manufactured by Toshiba

3.7.2.2 Planar-Cell Technology

INL and Cerametec have been testing stacks of 10-cm x 10-cm planar cells. Figure 3-5 shows the planar-cell concept and a 25-cell stack that was tested at temperatures ranging from 800 °C to 830 °C for 1000 hours in early 2006. As shown in Figure 3-6, the performance data showed a significant reduction in the hydrogen production rate over time, which probably is the result of degradation of cell seals. An SOE unit consisting of two 60-cell stacks was subsequently tested for 2040 hours. For this test, the hydrogen production rate dropped from 1.2 Nm^3/hr at the beginning of the test to 0.65 Nm^3/hr at the end of the test. Methodologies to improve sealing and long-term performance are currently being investigated. The next phase of the technology-development program is an integrated lab-scale test using an SOE unit consisting of four 60-cell stacks.



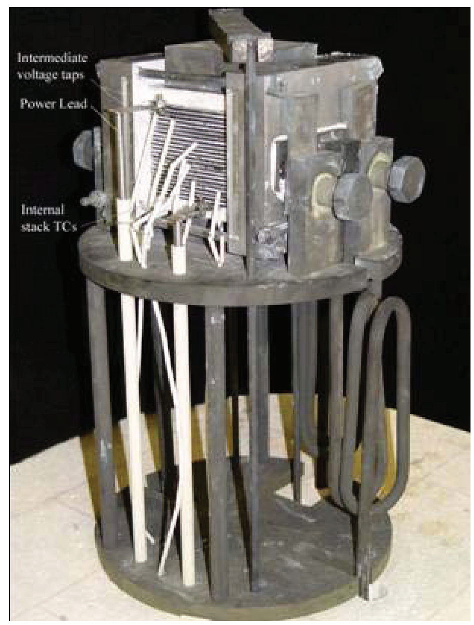


Figure 3-5. Planar-Cell Concept and 25-Cell Stack

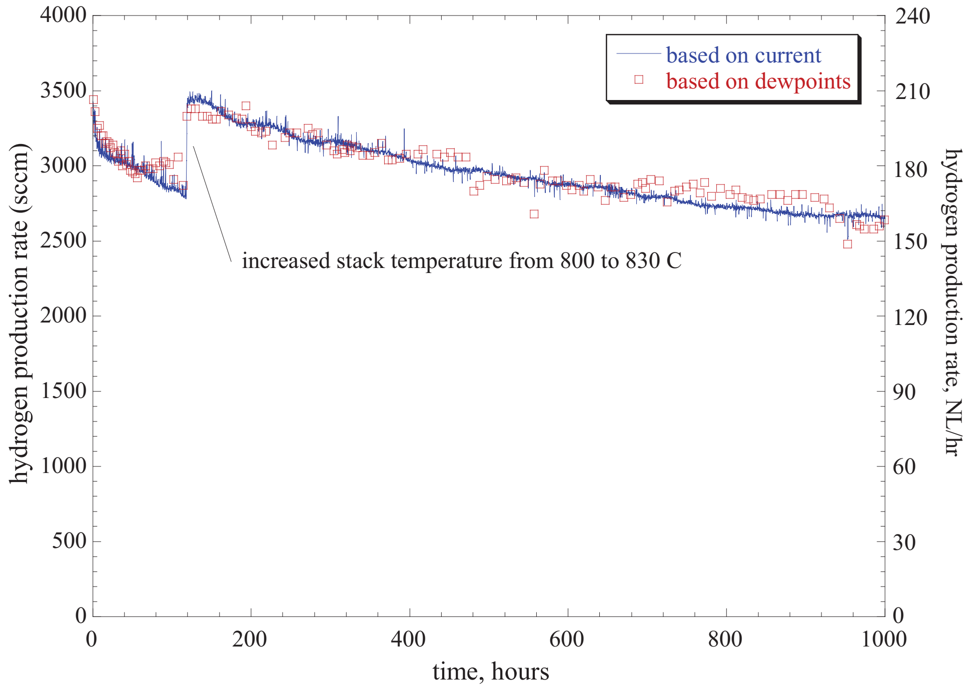


Figure 3-6. Performance Data from Testing of 25-Cell Stack

3.8 Molten-Salt Heat Transfer Media

Idaho National Laboratory has studied issues associated with the secondary heat transport system and possible design configuration using either high-pressure helium or molten salt as part of the NGNP project (Davis 2005), Recent INL focus has been on the Intermediate Heat Exchanger design in support of NGNP and the H2-MHR NERI project led by GA (Harvego 2006). The leading molten salt candidate is FLiNaK based on assessments prepared by ORNL (Williams 2006a, Williams 2006b) and studies being performed at University of California at Berkeley and University of Wisconsin (Petersen 2004). FLiNaK is a mixture of lithium fluoride, sodium fluoride, and potassium fluoride used in the Molten Salt Reactor Experiment at ORNL.

One of the most critical parameters for any molten salt is its melting (or freezing) temperature. For FLiNaK, the melting temperature is 454 °C. For a reactor outlet temperature of 950°C, the reactor inlet temperature could be as high as 590 °C. Under these conditions, the minimum loop temperature would be as high as 565 °C which provides adequate margin to freeze for normal operating conditions. For lower reactor inlet temperatures, a molten salt with a lower melting temperature would be preferable in order to increase the margin to freezing. Alternate fluoride, chloride and fluoroborate molten salts have been considered. Some of these alternates have melting temperatures below 400 °C which is highly desirable. However, these alternates have poorer heat transfer characteristics and higher technical risks compared to FLiNaK (Williams 2006b).

High-temperature corrosion is an issue with molten salts that is being researched. This issue will be a concern with licensing and code qualification. Past experience with FLiNaK at ORNL led to the development of Hastelloy N. The NGNP and its secondary heat transport loop will operate at temperatures that will not allow the use of Hastelloy N as a structural material. For this reason, it has been proposed that the IHX be made of alloy 617 instead of Hastelloy N. Whether alloy 617 will have acceptable resistance to high-temperature corrosion has yet to be proven. The high Cr content of alloy 617 may be unsuitable without the benefit of a reducing environment.

The technical risks associated with molten salts relative to those associated with helium remain significant at the present time. The principal technical risks with molten salt are high-temperature corrosion, freezing protection and drainage of molten salt from the IHX and process heat exchangers (PHXs). Also, the pressure difference between the primary and secondary coolants significantly degrades the operating lifetime of a metallic IHX due to creep stress. While technical solutions appear possible, there does not appear to be a compelling reason to choose molten salt over high-pressure helium, particularly in view of the high-level NGNP Project requirement to use the lowest-risk technology consistent with satisfying the NGNP objectives.

3.9 Spent Fuel Disposal

The currently available data to predict the in-repository performance of spent MHR fuel and to predict near-field radionuclide source terms is summarized below.

3.9.1 Coated-Particle Fuel

The key barriers for the long-term containment of radionuclides in spent MHR fuel elements are the PyC and SiC coatings of the particles. This subsection addresses the mechanical stability and corrosion resistance of TRISO coatings and the leach rates from fuel kernels exposed as a result of coating failure in-reactor or in-repository.

3.9.1.1 Radionuclide Release from Exposed Kernels

There are no explicit data for leaching rates for the exposed UCO fuel kernels. Leaching tests have been run on LWR UO_2 fuel pellets (e.g., Gray 1992 and Steward 1994), but these materials may not be representative of the HTGR fuel kernels. German researchers at the FZJ (formerly KFA) have studied leaching of 10%-enriched, TRISO UO_2 with burnups up to 11% FIMA in brine (Brinkmann 1988 and Kirch 1990. (The Germans typically leach samples in brine because their high-level waste repository was in a salt dome formation at Gorleben.) The observed leach rates from exposed UO_2 kernels were significant. For example, when UO_2 kernels were leached in brine at 90 ^{o}C, 18% of Sb, 10% of Cs and 4% of U were leached out in 100 days. The other actinides and lanthanides leached at rates lower than U; these include Pu, Am, Ru, Cm, Ce, Np, and Eu, which leached at 1% to 3% in 100 days. Evidently, exposed kernels will provide little holdup over long times in a repository environment.

3.9.1.2 TRISO Coating Performance

There are extensive data which may be used to evaluate the relatively short-term (e.g., up to about 30 years) capability of coated particles to withstand the level of internal gas pressure which may be expected to develop in MHR spent fuel in a repository environment (e.g., McCardell 1992). There are also irradiation and heating data which directly demonstrate the capability of the coatings to withstand internal gas pressure as high as that expected in GT-MHR spent fuel, for durations of up to several years. However, real-time data which demonstrate that the TRISO coatings will withstand such pressure over time frames important to MHR spent fuel disposal (e.g., 10^4 years or more) are obviously not available.

There are limited data available which may be used to estimate the oxidation rate of PyC in air at expected repository temperatures. Some measurements (Moormann 1995) have been made of the oxidation of pyrolytic carbon in air at temperatures typical of reactor normal operating and accident conditions, which are much higher than those in a waste repository. The PyC oxidation rate determined from these data may be extrapolated to lower temperatures.

Studies were performed at ORNL on simulated high-level nuclear waste coated with PyC and SiC to improve waste containment relative to glassified waste forms (Stinton 1981). Using the technology developed for manufacturing HTGR coated fuel particles, microspheres of simulated Savannah River Plant waste were coated with PyC and SiC coatings to reduce leachability. Leach tests on PyC-coated samples were run at 90 ^{o}C for times up to 28 days. The analysis found no detectable release of Cs, U, Sr, Zr, etc., whereas readily measurable leach rates were obtained for the glassified waste forms.

The corrosion behavior of graphite, PyC, and SiC has also been investigated for the purpose of evaluating improved barriers for nuclear-waste isolation (Gray 1980 and Gray 1982). Measurements were made of graphite, PyC, and SiC corrosion in liquid deionized water at 200 to 300 ^{o}C in a pressurized autoclave. The results indicated that the leach rate of graphite in deionized water was more than a factor of 10^5 slower than the leach rate of waste glass.

There are limited data available which may be used to estimate the long-term oxidation of SiC coatings by air at temperatures expected to occur in a geologic repository. The oxidation rate of SiC in air has been measured at temperatures typical of reactor normal operating and accident conditions (Shiroky 1986 and Vaughn 1990), which are much higher than those expected to occur in a waste repository.

3.9.2 Compact Matrix Materials

Some measurements have been made of the oxidation of resin matrix in air, including data from the German HTR program, but these measurements have been made for temperatures representative of reactor normal operation and accident conditions (Moormann 1995), which are well above the temperatures expected in a waste repository. Corrosion rates of German thermosetting resin-derived matrix materials at the elevated temperatures characteristic of in-reactor service have been measured at KFA (Loenissen 1987); corrosion rates for the resin matrix under development by the AGR program are expected to be very similar.

3.9.3 Nuclear Graphite

The fuel-element graphite provides mechanical containment of the fuel compacts and is a barrier to RN release in a repository environment. Consequently, the corrosion behavior of graphite is of importance. The trace impurities in graphite (and in the compact matrix) must be controlled because neutron activation of certain impurities can produce long-lived radionuclides that are radiologically significant for repository timeframes (e.g., 5730-yr C-14, 301,000-yr Cl-36, etc.). These topics are addressed below.

3.9.3.1 Corrosion Behavior

There are limited data available which may be used to estimate the rate of oxidation by air of H-451 graphite at repository temperatures. Oxidation rates for H-327 and H-451 graphites by air have been measured at temperatures in the range of 375 to 850EC (Jensen 1973 and Fuller 1992, respectively), which is well above anticipated repository temperatures.

An important issue for the qualification of whole GT-MHR fuel elements for permanent disposal is the confirmation that nuclear graphite is a "noncombustible" material. The relevant federal regulation (10CFR60) does not provide quantitative criteria for determining combustibility; consequently, other criteria or standards must be sought. A common standard used to assess the combustibility of materials is ASTM E-136 "Standard Test Method for Behavior of Materials in a Vertical Tube Furnace at 750EC." A nonreference nuclear-grade graphite, designated TSX, has been so tested (Quapp 1986). No flaming was observed from any of the TSX test specimens, which is one of the ASTM E-136 criteria for noncombustibility. However, the temperature rise in the specimens slightly exceeded the 30EC temperature rise allowed by ASTM E-136.[10]

Experiments have been performed to measure corrosion of nonreference, nuclear-grade graphite in deionized water and air (Gray 1980 and Gray 1982). The water tests were performed at 200 to 300 ^{o}C in a pressurized autoclave. The results indicated both the water and air reaction rates showed an Arrhenius temperature dependence. At 300 ^{o}C, the graphite oxidation rate was 100x higher in deionized water than in air, suggesting that the reaction rate in water may be controlled by a catalytic process involving dissolved oxygen. The leach rate of graphite in deionized water was more than a factor of 10^5 slower than the leach rate of waste glass.

3.9.3.2 Chemical Impurities

The fuel product specifications and graphite material specifications for various HTGR applications limit the chemical impurities in core materials in several ways: (1) explicit limits on certain neutron poisons, especially boron, expressed in terms of total allowed boron equivalents; (2) explicit limits on certain chemical impurities, such as iron, and classes of chemical impurities, such as transition metals; and (3) nonspecific limits on "total ash." In response, nuclear graphites and compact matrix materials have been characterized by various analytical methods to demonstrate compliance with these specifications (e.g., the H-451 graphite used in two FSV fuel reload segments, Hoffman 1973). However, these determinations

[10] It is noteworthy that an analogous international standard for combustibility, ISO R 1182, allows a temperature rise of up to 50°.

typically have not achieved a complete mass balance, and the “ash” contents have not been fully characterized.

Carbon-14 is an activation product, which is produced in HTGR fuel elements primarily through the neutron activation of N-14, via an (n,p) reaction, which exists as an impurity in graphite. Nitrogen-14 is introduced during the manufacture of the fuel compacts and the graphite fuel blocks. The other mode of C-14 production is by the activation of C-13, which is a rare isotope that is naturally present in graphite. Carbon-14 is one of the key radionuclides of concern for release by groundwater leaching. The current database for C-14 inventories in nuclear fuel-element graphite is limited. The most extensive data set is probably that obtained for Peach Bottom 1 for which an attempt was made to estimate an overall C-14 mass balance for the reactor core (Wichner 1980); the apparent range of N-14 concentrations on PB graphite varied over two orders of magnitude, but an initial N-14 concentration well in excess of 100 ppm was inferred at certain core locations. More limited measurements were made of C-14 inventories in Fort St. Vrain fuel and test elements (Montgomery 1986). Based upon the measured C-14 contents and relatively crude estimates of the time-integrated neutron fluxes at the in-core sample locations, the inferred, initial N-14 content ranged from 10-30 ppm.

Experiments were also performed to measure C-14 leach rates from various irradiated graphites, including specimens from the German AVR, French Magnox, Hanford and British Magnox reactors to support decommissioning activities. The C-14 observed leaching rates varied over several orders of magnitude. The highest leaching rates were observed for the French Magnox graphite; the values ranged from 0.004/yr to 0.03/yr (Gray 1989). Intermediate values, ~$5x10^{-4}$/yr, came from Hanford and British Magnox graphites (Gray 1989 and White 1984, respectively). The British samples were also irradiated in a CO_2-cooled Magnox reactor. The lowest leaching rate, ~$1x10^{-5}$/yr, was for German fuel pebbles, irradiated in the He-cooled AVR (Zhang 1993).

Several tests have been performed to estimate the radionuclide leaching rates from nuclear-grade graphite. Prior to leaching, the samples were machined to remove the surface layer, so that the remaining radioactivity in the samples resulted almost entirely from nuclear reactions with impurities. Pretest activity measurements showed significant quantities of the following nuclides: H-3, C-14, Fe-55, Co-60, Ba-133, Cs-134, Eu-154, and Eu-154. Leaching tests were performed using both groundwater and seawater. The groundwater tests were performed at 25 ^{o}C and a pressure of 1 bar for an exposure time of 150 days. The following nuclides were detected in the leachate samples: H-3, C-14, Co-60, Ba-133, and Cs-134; activities of other nuclides were too low to be detected.

3.10 Design Methods Development and Validation

The design methods for analyzing prismatic HTGRs were first elaborated in a series of Licensing Topical Reports (LTRs) prepared in the 1970s in support of the Large HTGRs then being designed and licensed; these LTRs include: core nuclear design (Merrill 1973), core thermal design (Shenoy 1974), fuel performance (Smith 1974), fission gas release (Haire 1974) and fission metal release (Alberstein 1976). While these LTRs are now three decades old, they still provide useful insight into the basic methodology and are recommended to the reader. A number of status reports on the development of improved design methods and the various efforts to validate them by benchmarking and comparison with integral test data have also been prepared.

A brief summary status of the prismatic core design methods is presented below. Most of the design methods used for the analysis of the plant systems, structures and components and for the balance of plant are commercially available design tools, such as ANSYS, SINDA/FLUENT, RELAP5, Pro/E, MATHCAD, etc., and they will not be addressed here since there is a whole literature devoted to them.

3.10.1 Core Nuclear

GA has over 35 years of High-Temperature, Gas-Cooled Reactor (HTGR) design experience; and code packages specific to the HTGR environment have been developed, or were adapted, for reactor physics, fuel performance, and core mechanical design.

GA's reactor physics codes were developed from basic neutron transport and diffusion theory (e.g., Merrill 1973). These methods were adapted to high-temperature, graphite-moderated systems to allow the calculation of temperature-dependent graphite scattering kernels, and the development of fine group cross sections for graphite systems from point-wise data (i.e., ENDF/B, JEF, and JENDL data sets). Methods were also developed to generate broad-group cross sections for HTGRs (TRISO particle fuel, graphite block cores, etc.) from these fine group data sets. Monte Carlo methods are now also used, including the MCNP and Monteburns code packages. Models for calculating TRISO fuel performance and fission product release in core were developed based on operating reactor experience and the results of multiple fuel irradiation tests and incorporated into design codes such as SURVEY and TRAFIC-FD (Section 3.10.3). Thermo-hydraulic models for gas-cooled, graphite-moderated systems were also developed and codified (Section 3.10.2). Stress analysis was performed by commercially available methods such as the ANSYS code. The result was a complete code sequence for HTGR design that has been approved by the NRC for HTGR design, and that was used for both Peach Bottom Unit I and FSV reactor design. This code sequence has also been used for MHR design and development. The key interfaces between the core analysis codes are shown in Figure 3-7, and the basic codes used for design purposes are listed below.

<u>HTGR Physics Codes</u>:

- NJOY6: extracts and processes ENDF/B-6 nuclear data.
- MICROR: reformats NJOY-6 output for MICROX use.
- MICROX/MICROBURN: broad group cross section generation for TRISO particle cores.
- GAUGE: 2D depletion calculations.
- DIF3D & BURP: 3D depletion calculations.
- SORT3D: converts DIF3D data into SURVEY format.

<u>Fission Product and Fuel Performance codes</u>:

- SURVEY: fuel temperatures, fuel performance and fission gas analysis
- TRAFIC-FD: fission metal release analysis.
- SORS: fuel failure and fission product release analysis during accidents.
- TAC2D: thermal analysis during accidents.

Thermal-hydraulic codes:

- ANSYS: Thermal, fluid flow and structural analyses
- POKE: Simplified thermal/flow analysis for prismatic cores
- SINDA-FLUENT: Detailed thermo-hydraulic analysis.

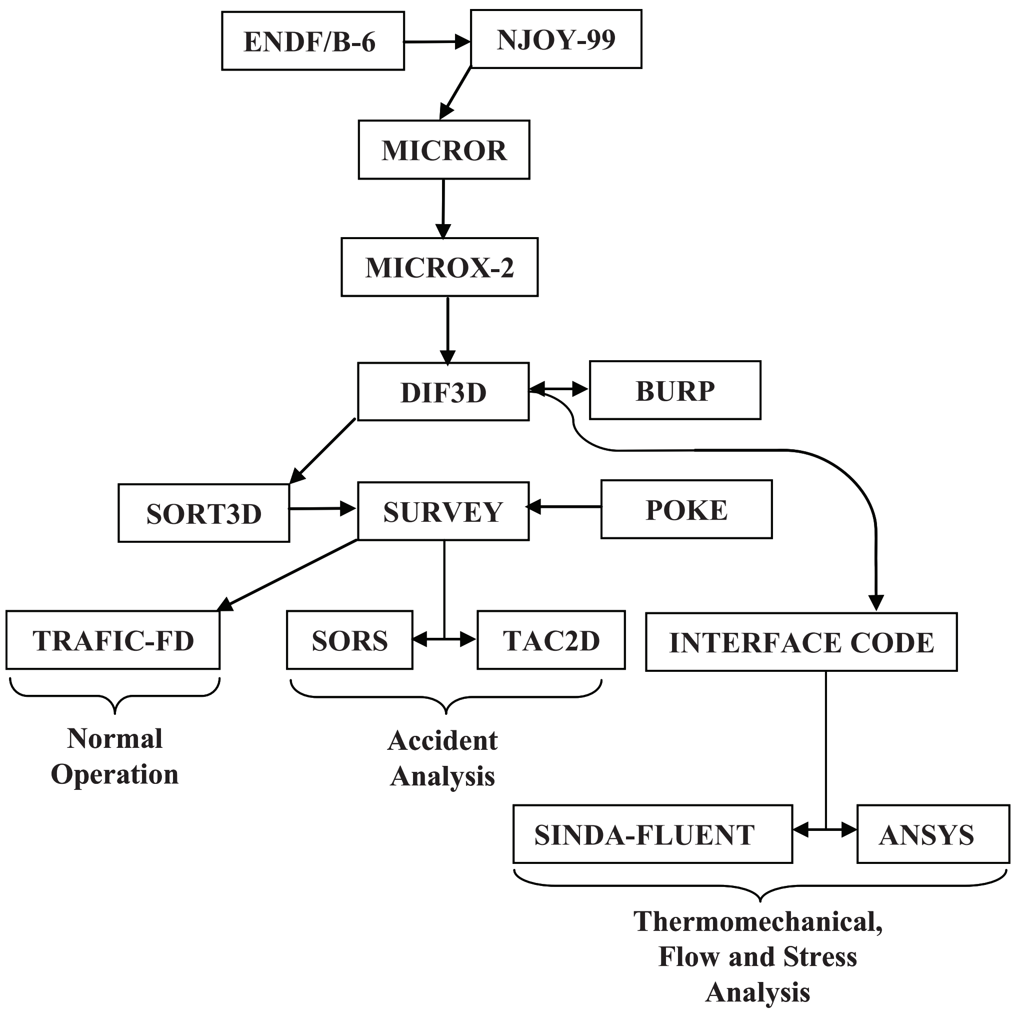


Figure 3-7. Key Interfaces between MHR Core Analysis Codes

The MHR core analysis codes listed above, and illustrated in Figure 3-7, are being integrated into an overall linked code package which will automate data transfer between the various components and provide a graphical input and output interface (GUI) to simplify the evaluation of design results. Modification of the codes is also planned to take advantage of the speed and computational capability of modern parallel processor computer clusters and super computers.

The ultimate goal is to provide design computation from first principles, which covers the full dimensional range of the MHR from TRISO particles to the complete core assembly.

The validation status of the core nuclear design methods is summarized in (PSID 1992). A reactor physics development plant was prepared for the commercial GT-MHR (Rucker 1992). The DDNs related to core nuclear design methods validation for the commercial GT-MHR (Section 5) appear to be adequate for the NGNP. However, a systemic review of the existing database used for nuclear code V&V needs to be performed, and feedback needs to be obtained from the NRC. While the experimental data used for nuclear code V&V are considered reliable, some of the older data and, in particular, the international data may not have an adequate QA pedigree to be accepted by the NRC without some confirmatory testing. Consequently, new DDNs may be identified during Conceptual or Preliminary Design. In particular reactivity worth calculations for control rods in the reflector of the annular core need to be validated to show that shutdown margins can be determined to the required accuracy under all conditions.

3.10.2 Thermal/Fluid Dynamics

The basic approach for performing core thermal/fluid flow analyses for prismatic HTGRs was established to support the design of Fort St. Vrain and the large HTGRs in the 1970s, and a number of codes were written at GA for that purpose (Shenoy 1974). While the analytical tools have evolved and the computational capabilities have improved enormously with modern computers, the basic analytical approach is still valid.

Future core thermal/flow analysis for normal operation and accidents will be performed with industry standard codes, such as ANSYS and RELAP5, and various commercial CFD codes as required. For example, Fuji Electric and GA have developed a 1/3 symmetry ANSYS model of the GT-MHR core (some results are discussed in the PCDSR). GA has prepared RELAP (ATHENA) models as part of the work with INL on the H2-MHR NERI project. Finally, GA has done some modeling of local core flow phenomena with the CFD code CFX (recently acquired by ANSYS) through collaborative work with KAERI.

3.10.3 Fuel Performance/Fission Product Transport

The design methods for predicting radionuclide source terms for design and safety analysis, including fuel performance and fission product transport throughout the reactor plant, were described in Section 3.2. Although these GA codes need modernization and formal validation (Hanson 2007a), they still represent a unique resource for the analysis of prismatic-core MHRs.

The validation status of these design methods was also described in Section 3.2. A verification and validation (V&V) plan for these codes was first prepared in 1988 (Maneke). These V&V plans were updated during the NPR program (e.g., Stone 1992). In particular, these V&V plans describe in detail the relationship between code development and validation and the planned fuel/fission product technology development programs: single-effects test data to upgrade the component models and material property correlations and independent integral test data to validate the upgraded codes. A revised V&V plan for these radionuclide control codes should be prepared for their application for NGNP design and licensing.

3.10.4 Core Structural Analysis

A number of core structural analysis codes were developed at GA during the past three decades and used extensively for core design and safety analysis. However, future core structural analysis, including seismic analysis, will be performed with ANSYS and ANSYS/DYNA3D. Improved constitutive equations for graphite along with improved material property data will be required (as defined in a series of DDNs).

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Table 3-1. Codes for Predicting Radionuclide Source Terms

Function	US Codes	Key Component Models/ Input Data	Prerequisites	Documentation
• Calculate radionuclide inventories in reactor core – Fission products – Transmutation products – Neutron activation products (including activation of impurities in core materials) • Used for zero-dimensional fuel cycle depletion, burnable poison loading analysis, and to compute long-term nuclide decay heat rates	GARGOYLE ORIGEN-2 (ORNL)	• Heavy-metal and graphite inventories • Impurities in core components • Core thermal power • Microscopic cross sections • Self-shielding nuclide coefficients • Core and region volumes • Bucklings	• Fuel-cycle characteristics • Fuel & graphite product specifications • Cross section libraries (e.g., ENDFB-6) • Multigroup energy structure • Nuclide depletion scheme (GARGOYLE) • Fission product yields (GARGOYLE)	GARGOYLE (CEGA-002922) ORIGEN-2 (ORNL-TM-7175)
• Calculate overall plant mass balance for radionuclides - Total core/spent fuel elements - Total circulating activities - Total plateout inventories - Purification system inventories • Used to generate radionuclide design criteria	RADC RANDI	• Isotope nuclear properties (e.g., fission yields, decay constants, decay chains, etc.) • He mass flow rate • Total circulating He inventory • He purification rate	• Basic plant design parameters (e.g., power level, plant lifetime, capacity factor, etc.) • Basic core design parameters (e.g., number of fuel elements, fuel residence time, etc.,)	Meek and Rider, “Compilation of Fission Product Yields,” 1978 Lederer, “Table of Isotopes,” 1978 RADC manual (CEGA-002814) RANDI manual (GA-A14091)
• Calculate thermal and stress histories for TRISO-coated fuel particles • Used to specify particle attributes (e.g., kernel diameter, coating thicknesses, etc.) for fuel particles	PISA SOLGASMIX-PV	• Material properties of pyrocarbon and SiC coatings as function of temperature and fast fluence • Thermal properties of pyrocarbon and SiC coatings • Total yields of fission gases as function of burnup.	• Core operating envelope (burnup, fast fluence, temperature) • As-manufactured fuel attributes (e.g., allowable standard deviations in kernel and coating dimensions, allowable coating defects, etc.) • Allowable failure during normal operation and accidents	Material Properties (CEGA-002820) FUEL manual (EG&G report) PISA manual (CEGA-002550) FDDM[11] (GA 901866/F)

[11] Fuel Design Data Manual (FDDM, GA Proprietary Data) contains and controls the reference models and material property data used for fuel performance and fission product transport analyses at GA.

Function	US Codes	Key Component Models/ Input Data	Prerequisites	Documentation
• Calculate 3-D fuel performance analysis for normal operation - Fuel failure - Fission gas release • To determine whether core design meets radionuclide design criteria	SURVEY	• Fuel particle design • As-manufactured fuel attributes • Fuel particle performance models • Fission gas release models	• 3-D power distributions (DIF3D/SORT3D output) • Core operating envelope • Flow distribution • Radionuclide design criteria to define allowable core releases	GA-LTRs-14/-15/-17 (dated[12]) SURVEY manual (CEGA-002927, UCNI) FDDM (GA 901866/F) PC-MHR analysis
• Calculate 3-D fission metal release for normal operation - Release from kernels - Release from particles - Release from fuel element • To determine whether core design meets radionuclide design criteria	COPAR-FD TRAFIC-FD	• Fuel particle design • As-manufactured fuel attributes • Diffusion coefficients (kernels, coatings, graphite) • Sorption isotherms for matrix and graphite	• 3-D power, burnup & fluence distributions (DIF3/SORT3D output) • Fuel failure distributions (SURVEY output) • Radionuclide design criteria to define allowable core releases	GA-LTR-20 (dated[3]) COPAR-FD manual (CEGA-002098) TRAFIC-FD manual (CEGA-001904) FDDM (GA 901866/F) PC-MHR analysis
• Calculate plateout distributions in primary coolant circuit • To provide source terms for: - Calculating O&M dose rates - Shielding & cask design - Depressurization accidents ("liftoff")	PADLOC	• Convective mass transfer correlations [Sh = Sh(Re, Sc)] • Sorption isotherms for graphite and primary circuit (PC) metals	• PC conceptual design (e.g., geometry, materials of construction, etc.) • PC operating conditions (pressure, temperature, flow rate) • Radionuclide design criteria to	PADLOC manual (GA-A14401) FDDM (GA 901866/F) PC-MHR analysis

[12] Portions of these Licensing Topical Reports (LTRs), which were prepared in the 1970s, are obsolete, but the fundamental approach and basic analytical methodologies described therein are still applicable.

Function	US Codes	Key Component Models/ Input Data	Prerequisites	Documentation
			provide total plateout inventories	
• Calculate production rates and overall plant mass balance for tritium[13] - Fuel elements - Reflector elements - Total circulating inventory - Permeation through heat exchanger to contaminate secondary coolants • Used to generate H-3 source terms: - Normal environmental discharges - Disposal of spent fuel elements - Disposal of spent reflector elements	TRITGO	• He-3/He-total in primary He • Li impurities in core materials (from product specifications) • Sorption isotherms for matrix and core graphites • Permeation coefficient correlations for PC metals	• RS/PC conceptual design (e.g., geometry, materials of construction, etc.) • RS/PC operating conditions (pressure, temperature, flow rate) • Plant primary coolant chemistry specification (i.e., allowable coolant impurities, especially H_2O and H_2) • He purification system flow rate and efficiency for H-3 removal	ORNL-TM-4303 TRITGO manual (GA 911081/0) FDDM (GA 901866/F) MHTGR (steam cycle) analysis
• Calculate production rates and overall plant mass balance for Carbon-14 - Fuel elements - Reflector elements • Generate C-14 source terms: - Disposal of spent fuel elements - Disposal of spent reflector elements	GARGOYLE None (hand calculations)	• N-14 impurities in core materials (from product specifications) • Nuclear properties (e.g., cross sections)	• RS conceptual design (e.g., geometry, materials of construction, etc.) • Thermal flux distributions in reactor core	GARGOYLE (CEGA-002922) Nuclear Engineering Handbooks MHTGR (steam cycle) analysis

[13] Importance of tritium in a direct-cycle GT-MHR is to be determined; should be less of significance in GT-MHR than in a steam-cycle plant (e.g., MHTGR).

Function	US Codes	Key Component Models/ Input Data	Prerequisites	Documentation
• Calculate 3-D fuel performance and fission product release for core heatup accidents - Incremental fuel failure - Fission gas release - Fission metal release • To determine whether design meets accident dose limits	SORS/NP1	• Fuel particle design • As-manufactured fuel attributes • Fuel performance models for accident conditions • Diffusion coefficients (kernels, coatings, graphite) • Sorption isotherms for matrix and graphite	• Definition of Design Basis Accidents (DBAs) and Beyond-Design Basis Accidents (BDBAs) • Initial conditions from core analyses for normal operation (e.g., power distributions, etc.) • 3-D, transient thermal/hydraulic performance of core during accident	GA-A15439 (dated[2]) SORS/NP1 manual (CEGA-002092) FDDM (GA 901866/F) MHTGR (steam cycle) analysis
• Calculate 3-D fuel performance analysis for water ingress accidents[14] - Incremental fuel failure - Fission gas release (especially I isotopes) - Fission metal release - Core corrosion • To determine whether core design meets accident dose limits	OXIDE-4	• Fuel particle design • As-manufactured fuel attributes • Reaction kinetics for H_2O and core materials • Fuel performance models for H_2O ingress conditions • Diffusion coefficients (kernels, coatings, graphite) • Sorption isotherms for matrix and graphite	• Definition of Design Basis Accidents and Beyond-Design Basis Accidents • Initial conditions from core analyses for normal operation (e.g., power distributions, fuel failure distributions, etc.) • Transient H_2O ingress rates • 3-D, transient thermal/hydraulic performance of core during accident • Definition of allowable core release rates during core heatup accidents (derived from allowable offsite doses).	OXIDE4 manual (CEGA-001871) FDDM (GA 901866/F) MHTGR (steam cycle) analysis
• Calculate fractional reentrainment ("liftoff") of plateout activity in PC during depressurization accidents - Source term for offsite dose - Contamination of reactor containment building • To determine whether plant design meets accident dose limits	POLO	• Liftoff correlations (fractional reentrainment as function of shear ratio[15])	• Definition of Design Basis Accidents and Beyond-Design Basis Accidents • PC conceptual design (e.g., geometry, materials of construction, etc.) • PC operating conditions during depressurization transient (to calculate shear ratios) • Initial plateout distributions (EOL plateout inventories assumed)	POLO manual (DOE-HTGR-88332) FDDM (GA 901866/F) MHTGR analysis

[14]Importance of water-ingress accidents in a direct-cycle GT-MHR is to be determined; such accidents should be of much less significance in a GT-MHR than in a steam-cycle plant (e.g., MHTGR).
[15] Shear ratio is the ratio of the wall shear stress during a depressurization transient to the wall shear stress during normal 100% power operation.

Function	US Codes	Key Component Models/ Input Data	Prerequisites	Documentation
• Calculate radionuclide transport from reactor to site boundary during DBAs and BDBAs - Radionuclide retention in reactor containment building - Radionuclide transport in atmosphere (plume dispersion) - On-site attenuation (e.g., fallout, washout) • To determine whether plant design meets site-boundary dose limits • To determine whether plant design meets risk goals.	MACCS (SNL)	• Physical and chemical forms of radionuclides released from PC (e.g., aerosols, I speciation, etc.) • Radionuclide attenuation models (e.g., settling, deposition, condensation, etc.)	• Definition of Design Basis Accidents and Beyond-Design Basis Accidents • Transient radionuclide release rates from PC to containment (e.g., SORS/NP1 or OXIDE4 output) • Environmental conditions in containment building • Plant site characteristics (e.g., physical dimensions, meteorology, demographics, etc.) • Definition of allowable offsite doses for DBAs and BDBAs.	MACCS reports/users manual (NUREG/CR-4691) MHTGR analysis

4. DESIGN REQUIREMENTS

The design requirements imposed upon the NGNP will ultimately determine what technology is needed to support plant design and licensing. Consequently, determination of the design requirements is a prerequisite to defining the Design Data Needs and attendant technology development programs for the NGNP. In fact, the fundamental reason that the current NGNP and NHI R&D programs lack focus is that they are, in general, generic programs that have not been scoped or prioritized to support a particular plant design. Since the NGNP is still the preconceptual design phase, certain design requirements are provisional, especially the lower-level ones; consequently, the conclusions presented in this TDP regarding the current R&D programs are subject to revision as the design matures and more definitive feedback is provided by regulators and potential customers.

As described in Section 3, there is a large, often robust, international database to support most aspects of NGNP design as a result of five decades of nuclear plant design and operation, especially the design and operation of seven HTGRs (Section 2.3.3). Consequently, most design requirements do not generate DDNs and can be satisfied by standard engineering practice and application of validated analytical tools. In fact, a relatively few design requirements generate most of the DDNs that have been identified for the NGNP at this time; those requirements which, in large measure, drive the technology development requirements are highlighted in this section.

4.1 Provisional NGNP Design Requirements

The System Requirements Manual (SRM) is intended to be the top-level design document for the NGNP. The SRM serves as the roadmap document that identifies the source of the NGNP top-level requirements (i.e., mission needs and objectives) and how these top-level requirements flow down through subordinate requirements at the plant, system, subsystem, component, and ultimately the part level. Design requirements for the NGNP include both institutionally imposed and functionally derived requirements. Each preconceptual engineering services contractor is preparing an SRM as part of its workscope.

The System Requirements Manual prepared by GA (SRM 2007) has adopted a particular protocol for identifying requirements which is reproduced here:

"If the plant-level requirement is an institutional requirement, the source of the requirement is given in brackets following the requirement. If a source is not shown following the statement of the requirement, the requirement is a functionally derived requirement. A number is assigned to each requirement for identification purposes. The identification number has the format 3.x.y where 3.x is the SRM section number and y is the requirement number. If a requirement is subordinate to a higher-level requirement (i.e., it stems from the higher-level requirement), the subordinate requirement has the format 3.x.y.z, where 3.x.y is the identification number for the higher-level requirement and z is the unique number for the subordinate requirement. Brackets { } are used herein to identify a value that is preliminary in nature because of design uncertainty or insufficient documentation, or that requires verification."

The plant-level and system-level requirements given in the GA-prepared SRM have been reviewed, and those judged to generate significant DDNs are reproduced in the following section; lower-level requirements (e.g., component-level) have not yet been formally derived. The various references cited in the requirements are identified in Table 4-1. In some cases, the decision to include or exclude a certain requirement was rather arbitrary; in any case, the plant design will have to meet all of the requirements whether they are included here or not.

Table 4-1. SRM References Cited in Requirements

Ref. 2	"Next Generation Nuclear Plant – High Level Functions And Requirements," Ineel/Ext-03-01163, Idaho National Laboratory, September 2003
Ref. 9	"Utility/User Incentives, Policies, and Requirements for the Gas Turbine-Modular Helium Reactor," DOE-GT-MHR-100248, Rev. 0, Technology Insights, September 1995

4.2 Key Requirements Driving Technology Development

The plant-level and system-level requirements judged to generate significant DDNs are reproduced in this section where they are grouped by major technology development areas. The programmatic implications of these requirements are discussed, and in some cases provisional lower-level design criteria are proposed.

4.2.1 User/Utility Radionuclide Control Requirements

4.2.1.1 SRM Requirements

PLT 3.1.1.2 - The NGNP reactor shall have a prismatic block core.

PLT 3.1.1.3 – The NGNP shall use qualified TRISO-coated uranium oxycarbide (UCO) or uranium dioxide fuel. The fuel particles shall be agglomerated into cylindrical compacts. Qualified uranium dioxide fuel may be acceptable for initial fuel loading, but shall be replaced by UCO, when it is has been qualified. [Ref. 2, Sections 3.1.7 and 3.1.10]

PLT 3.1.8 – The NGNP shall be designed to achieve fuel burn up consistent with maximum fuel utilization while minimizing waste streams, optimizing fuel economics, and ensuring low proliferation risk. [Ref. 2, Section. 3.1.9].

PLT 3.1.9 - The NGNP shall be designed to satisfy the following top-level radionuclide control regulatory requirements:

- During normal operation, offsite radiation doses to the public shall be < limits specified in Appendix I of 10 CFR 50 and 40 CFR 190
- Occupational radiation exposures shall be ≤10% of the limits specified in 10 CFR 20
- During DBAs, offsite doses at the site EAB shall be less than those specified in the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA-520/1-75-001) for sheltering and evacuation

[Ref. 9, Section 3.1.13 and U/U Requirement, SRM Section 2.3.5, Fig. 1].

PLT 3.1.10 - The design of the NGNP systems and processes shall be such that the volume of low-level radioactive dry and wet waste, as shipped off-site, shall be less than 3.6 m^3, annually (excluding replaceable reflector elements). [U/U Requirement, SRM Section 2.3.5, Fig. 1].

PLT 3.1.11 - Qualified fuel, including fuel product and fuel fabrication specifications, a QA plan, demonstrated irradiation performance and fuel performance codes to predict fuel performance as a function of operating condition.

PLT 3.1.11.6 - The NGNP shall be designed to demonstrate a probability of < 5 x 10^{-7} per plant year that offsite doses at or beyond the site EAB of 425 meters will [not] exceed the limits specified in the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA-520/1-75-001) for sheltering and evacuation. [U/U Requirement, SRM Section 2.3.5, Fig. 1].

PLT 3.1.11.7 - The NGNP shall be designed to demonstrate that plant personnel exposure of <70 person-rem/GW_e-year. [U/U Requirement, SRM Section 2.3.5, Fig. 1].

PLT 3.4.5 - The interface system between the NGNP and the hydrogen production plants shall be designed to ensure that tritium migration into the hydrogen production systems will be limited, such that the maximum amount of tritium released from the integrated NGNP facilities or found in drinking water does not exceed EPA standards. [Ref. 2, Section 3.4.5].

PLT 3.5.8 – For demonstration of commercial plant radiological source terms, the NGNP shall be designed to experimentally determine the fission product activity that could potentially be released should there be a rupture in the primary coolant boundary. [PLT 3.1.9; PLT 3.1.11.6].

The reference fuel cycle shall be based on the use of a once-through uranium fuel cycle with U-235 enrichment no greater than 19.9%. (System 11)[16]

4.2.1.2 Programmatic Implementation

The above radionuclide control requirements mandated for the NGNP by the SRM (2007) are essentially the same requirements imposed previously on the commercial GT-MHR (Shenoy 1996) and, prior to that, on the steam-cycle MHTGR (PSID 1992). Stringent limits on fuel performance and as-manufactured fuel quality will be necessary to meet these top-level radionuclide control requirements. The challenge has become significantly greater because the core outlet temperature has been progressively increased from 700 °C for the steam-cycle MHTGR to 850 °C for the direct-cycle GT-MHR to 950 °C for the NGNP, and the core power (hence, the core radionuclide inventories) has been increased from 350 to 600 MW(t).

In addition, these top-level radionuclide control requirements mandate the upgrading and validation of the design methods used to predict radionuclide source terms for plant design and licensing.

4.2.1.2.1 Fuel Requirements

Fuel performance requirements have not been formally adopted for the NGNP, but provisional requirements have been recommended for a generic VHTR (Hanson 2004); they are reproduced below as an indication of the fuel requirements that should be anticipated for the NGNP.

The logic for deriving these fuel requirements is illustrated in Figure 4-1 (Hanson 2001). Top-level requirements for the VHTR will be defined by both the regulators and the user. Lower-level requirements will then be systematically derived using a top-down functional analysis

[16] The system-level requirements in the SRM are not numbered at this writing.

methodology. With this approach, the radionuclide control requirements for each of the release barriers can be defined. For example, starting with the allowable doses at the site boundary, limits on Curie releases from the VLPC, from the reactor vessel, and from the reactor core will be successively derived. Fuel failure criteria will in turn be derived from the allowable core release limits. Finally, the required as-manufactured fuel attributes will be derived from the in-reactor fuel failure criteria providing a logical basis for the fuel quality specifications.

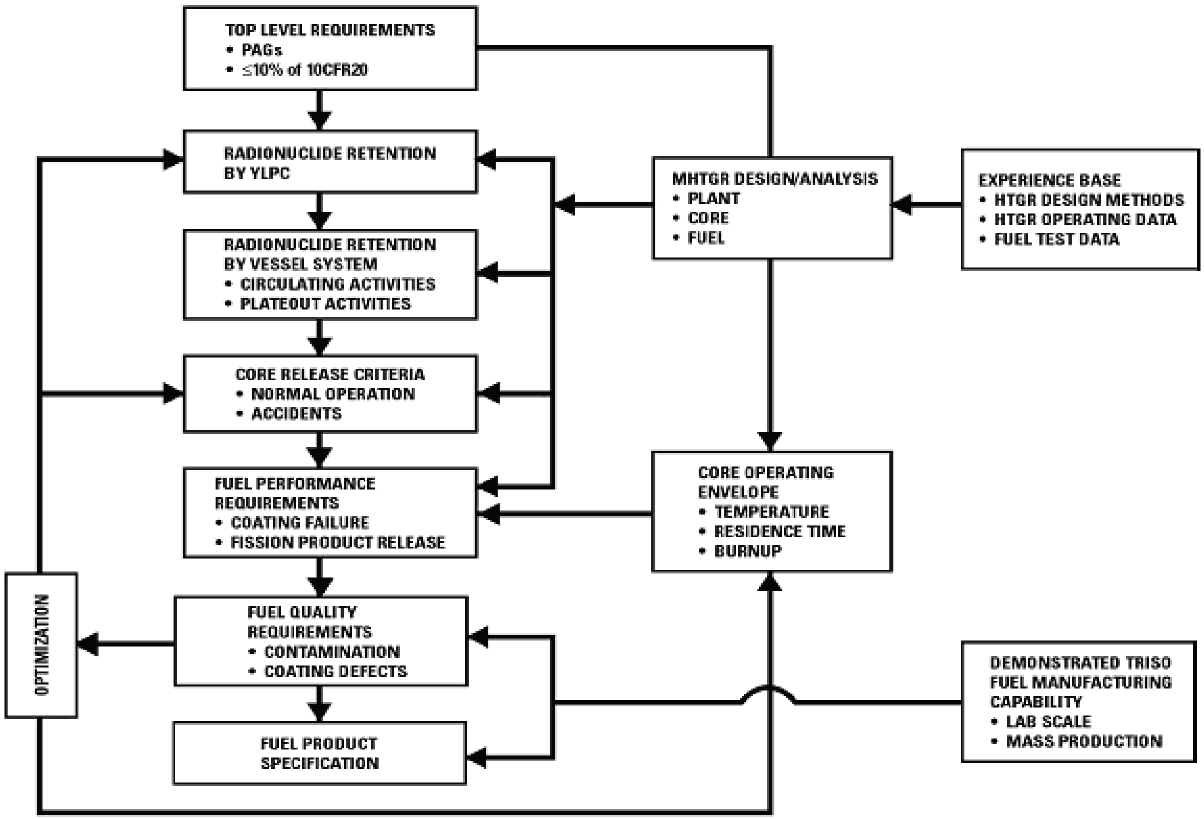

Figure 4-1. Logic for Deriving Fuel Requirements

In-service fuel performance requirements and as-manufactured fuel quality requirements have not yet been defined for a generic VHTR or for the NGNP. The fuel performance and quality requirements adopted for a given HTGR design along with the fuel service conditions will determine the amount of technology development that will be necessary to support the design and licensing of the plant. Consequently, it is critically important that a comprehensive set of fuel requirements be derived for the NGNP early in the design process.

As a point of departure for preparing this TDP, the fuel requirements for the VHTR with a 950 °C core outlet temperature were assumed to be the same as those for the direct-cycle GT-MHR with a 850 °C core outlet temperature (Munoz 1994). This assumption may prove to be too ambitious. It is reasonable to expect that these as-manufactured fuel quality limits can be met since the Germans met or exceeded comparable limits in the late 1970s (e.g., Hanson 2001). However, the in-service fuel performance limits could prove problematic; in particular, the allowable core metal release limits (Ag, Cs, etc.) may have to be increased even if the failure limits are maintained because of the higher average core temperatures which will result in less overall retention by the fuel kernels of failed particles and by the fuel-element graphite.

The provisional NGNP fuel performance and quality requirements are summarized in Table 4-2, and the provisional metal release limits are shown in Table 4-3. For perspective, the allowable metal release limits for the US steam-cycle MHTGR plant and for the German direct-cycle HHT plant are also shown in the latter table. The NGNP limits on volatile metal release are particularly speculative at this writing (because they were developed for a direct-cycle GT-MHR rather than for a VHTR), and considerable plant design and fuel development will likely be required to optimize them.

Table 4-2. Provisional Fuel Requirements for NGNP

Parameter	Commercial GT-MHR		VHTR	
	≥50% Confidence	≥95% Confidence	≥50% Confidence	≥95% Confidence
As-Manufactured Fuel Quality				
Missing or defective buffer	≤1.0 x 10^{-5}	≤2.0 x 10^{-5}	[≤1.0 x 10^{-5}]	[≤2.0 x 10^{-5}]
Defective SiC	≤5.0 x 10^{-5}	≤1.0 x 10^{-4}	[≤5.0 x 10^{-5}]	[≤1.0 x 10^{-4}]
Heavy metal (HM) contamination	≤1.0 x 10^{-5}	≤2.0 x 10^{-5}	[≤1.0 x 10^{-5}]	[≤2.0 x 10^{-5}]
Total fraction HM outside intact SiC	≤6.0 x 10^{-5}	≤1.2 x 10^{-4}	[≤6.0 x 10^{-5}]	[≤1.2 x 10^{-4}]
In-Service Fuel Performance				
Normal operation	≤5.0 x 10^{-5}	≤2.0 x 10^{-4}	[≤1.0 x 10^{-4}]	[≤4.0 x 10^{-4}]
Core heatup accidents	[≤1.5 x 10^{-4}][(a)]	[≤6.0 x 10^{-4}]	[≤3.0 x 10^{-4}]	[≤1.2 x 10^{-3}]

[(a)]Values in [square brackets] are provisional and subject to revision as the design evolves.

Table 4-3. Provisional Metal Release Limits for NGNP

Reactor Plant	Type	COT[(a)] (°C)	Allowable Core Fractional Release			
			Cs-137		Ag-110m	
			"Maximum Expected"	"Design"	"Maximum Expected"	"Design"
MHTGR	Steam-cycle	700	≤7.0 x 10^{-6}	≤7.0 x 10^{-5}	≤5.0 x 10^{-4}	≤5.0 x 10^{-3}
HHT	Direct-cycle	850	≤2.0 x 10^{-5}	≤1.0 x 10^{-4}	≤8.6 x 10^{-5}	≤6.5 x 10^{-4}
GT-MHR	Direct-cycle	850	≤1.0 x 10^{-5}	≤1.0 x 10^{-4}	≤2.0 x 10^{-4}	≤2.0 x 10^{-3}
VHTR	Process heat	950	≤ [1.0 x 10^{-5}]	≤ [1.0 x 10^{-4}]	≤[2.0 x 10^{-4}]	≤[2.0 x 10^{-3}]

[(a)]COT = core outlet temperature

The provisional requirements given in Tables 4-2 and 4-3 reflect the standard GA design practice of defining a two-tier set of radionuclide design criteria - referred to as "Maximum Expected" and "Design" criteria - or allowable core releases for normal operation and Anticipated Operational Occurrences (Hanson 2004); this practice has been followed since the design of the Peach Bottom 1 prototype US HTGR up through the current commercial GT-MHR. The "Design" criteria are derived from externally imposed requirements, such as the site-boundary dose limits, occupational exposure limits, etc.; in principle, any of these radionuclide control requirements could be the most constraining for a given reactor design.

Once the "Design" criteria have been derived from the radionuclide control requirements, the corresponding "Maximum Expected," criteria are derived by dividing the "Design" criteria by an uncertainty factor, or design margin, to account for uncertainties in the design methods and reactor service conditions. This uncertainty factor is typically a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals. The fuel and core are to be designed such that there is at least a 50% probability that the fission product release will be less than the "Maximum Expected" criteria and at least a 95% probability that the release will be less than the "Design" criteria.

This GA approach to implementing such radionuclide design criteria is illustrated in Figure 4-2. (No particular scale is implied in this figure; it is simply a conceptual illustration of the approach.) In the example given in the figure, the preliminary design predictions (solid lines) slightly exceed the criteria (triple lines) at the 50% confidence level: i.e., the nominal (50% confident) prediction is slightly higher than the "Maximum Expected" criterion, but the 95% confident prediction meets the "Design" criterion, primarily because a large design margin was chosen to accommodate the considerable uncertainties in the current design methods at the preliminary design stage. This example was chosen because it is anticipated to roughly reflect the current prediction of Ag-110m release from a commercial GT-MHR core, based upon previous GA analysis of a WPu-burning core operating with an 850 ^{o}C core outlet temperature. Silver release is of concern because it can be diffusively released from intact TRISO particles at high temperatures and preferentially deposit on the turbine, where it is predicted to be a dominant contributor to O&M dose rates (it is only a minor contributor to offsite dose rates because of its low effectivity).[17]

There are several candidate options for resolving this design issue. The first option is simply to relax the "Maximum Expected" criterion and to design the plant to accommodate the currently predicted levels of Ag release and the large uncertainties in the predictive methods; however, this option implies high O&M dose rates and the attendant requirements for fully remote turbine maintenance, etc. Another option is to develop and qualify efficient decontamination protocols to reduce the dose rates from the turbine prior to refurbishment to levels permitting hands-on maintenance. A third option (dashed lines) is to reduce the predicted Ag release and the uncertainties therein by a combination of design optimization (primarily to reduce the nominal prediction) and technology development (primarily to reduce the uncertainty in the prediction).

[17] Silver release and plateout will be somewhat less of an issue in a commercial GT-MHR because less Ag-110m is produced with LEU fuel compared to WPu fuel, but even with LEU fuel it is still predicted to a major contributor to O&M dose rates, along with Cs-134 and Cs-137.

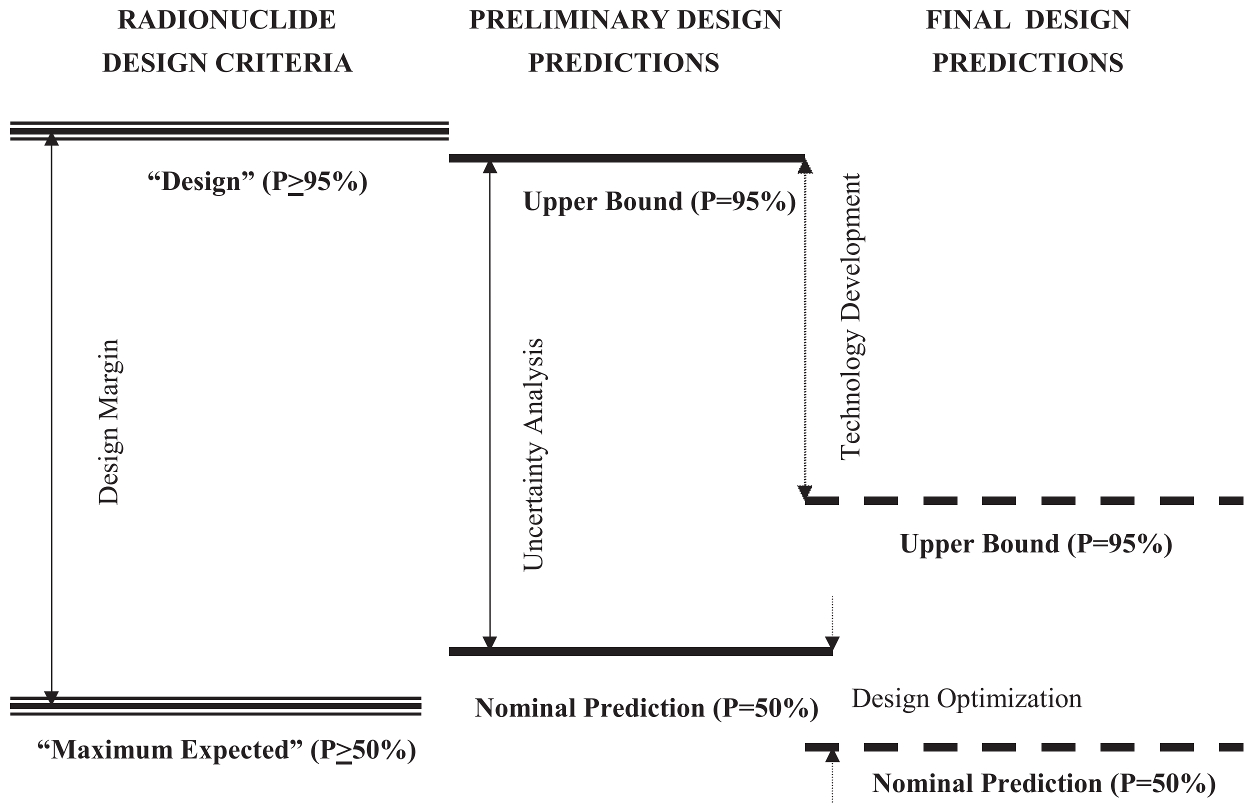

Figure 4-2. Radionuclide Design Criteria

Since diffusive release from intact particles is the dominant source of Ag release, the most effective design changes to reduce Ag release are those that reduce the peak fuel temperatures in the core. Some reduction in peak temperatures can be achieved by improved fuel zoning to optimize the core power distribution for minimum Ag release, and further reductions are possible with various fuel shuffling schemes. Larger fuel temperature reductions require more dramatic changes in the fuel-block design and/or in core operating conditions (e.g., power density); such changes have broad implications for the overall plant design and fuel cycle costs.

A comprehensive trade study would be required to identify the optimal combination of the above options to resolve the Ag plateout issue. In any case, it would be prudent to design a first-of-a-kind, direct-cycle HTGR to permit fully remote turbine maintenance should the actual gamma dose rates prove to be higher than predicted.

The above discussion of radionuclide design criteria also serves to illustrate a fundamental construct with regard to the required predictive accuracies of the design methods used for the design and safety analysis of an HTGR. All design methods do not *a priori* have to be highly accurate; however, there must be sufficient design margin to reliably account for the uncertainties in the predictions. In some cases, it has proven impractical or uneconomical to add large design margins; consequently, the design methods for such applications are required to be highly accurate (the nuclear design methods for predicting core reactivity are an example).

In other cases, an order-of-magnitude uncertainty can be tolerated by adding sufficient margin (an example is specifying thicker shielding for a component with a complex geometry for which radioactive source term is not well known prior to actual service).

4.2.1.2.2 Validated Source Terms

As described previously, the radionuclide containment system for the NGNP will be comprised of multiple barriers to limit RN release from the core to the environment to insignificant levels during normal operation and a spectrum of postulated accidents. To reiterate, the five principal release barriers are: (1) the fuel kernel, (2) the particle coatings, particularly the SiC coating, (3) the fuel-element structural graphite, (4) the primary coolant pressure boundary; and (5) the Vented Low-Pressure Confinement building. As part of the design process, performance requirements must be derived for each of these release barriers.

When the fuel requirements presented in the previous section were derived, credit was taken for radionuclide retention by each of the release barriers. Barrier performance requirements are specified such that only the particle coatings are needed to meet 10CFR100 dose limits; however, credit for the additional barriers is taken to meet the EPA Protective Action Guide (PAG) dose limits (Hanson 2001). The alternative would be to set fuel failure limits sufficiently low that the PAG dose limits could be met even if it were assumed that 100% of the fission product inventories of failed particles were released to the environment. This approach is considered impractical. For perspective, for the 350 MW(t) steam-cycle MHTGR, the allowable I-131 release limits to meet the PAG thyroid dose limit of 5 rem were 2.6 Ci for short-term events, such a rapid depressurization, and 29 Ci for long-term events, such as a depressurized core conduction cooldown (Hanson 2001). Converting these Curie limits to allowable fuel failure fractions for a 600 MW(t) GT-MHR gives limits of ~10^{-7} during normal operation and ~10^{-6} during core heatup events, respectively.

The irradiation and postirradiation programs to justify such low failure fractions for TRISO fuel would be massive, and the QC costs to verify the corresponding as-manufactured fuel quality requirements during mass production would likely be prohibitively expensive. Consequently, credit is taken for iodine retention in the kernels of failed particles (10x)[18] and for iodine retention in the VLPC (10x) during core heatup accidents. If no credit is taken for retention in the VLPC, the allowable fuel failure fraction would have to be reduced by an order of magnitude (to ~10^{-6} during normal operation and ~10^{-5} during core heatup accidents).

Once again, validated analytical tools are needed to quantify the performance of each of the RN release barriers. The fuel product specifications developed using the above protocols are only as reliable as the analysis methods used in their derivation

[18] This assumption that the core-average, fractional release of I-131 from exposed kernels is only 10% is highly uncertain. If higher release fractions are determined experimentally for UCO kernels, then there will be an incentive to take even more credit for holdup in the VLPC.

4.2.2 Operation at 950 °C Core Outlet Temperature

4.2.2.1 SRM Requirements

PLT 3.1.1.3 - The NGNP reactor system shall be designed to operate with an average core outlet coolant temperature range of 850°C to 950°C. [Ref. 1, Section 3.3]

PLT 3.1.4 – Average reactor outlet temperature in the range 850ºC to 950ºC, with future capability to increase it to above 1000°C

The Primary Heat Transport System shall be designed to operate at a nominal reactor coolant outlet temperature in the range of 850 °C to 950 °C. The system shall be capable of transporting helium primary coolant from the reactor core outlet plenum to the IHX, and from the IHX to the reactor core inlet plenum (System 13).

The PCS turbomachine shall utilize a coupled generator and turbomachine rotor design and shall be designed to operate continuously at a nominal turbine inlet temperature of 950 °C (System 41).

The PCS turbomachine and generator rotors shall be supported by magnetic bearings (System 41).

The PCS recuperator heat exchanger shall be designed to achieve a minimum effectiveness of {95%} (System 41).

4.2.2.2 Programmatic Implementation

The Reactor System, Power Conversion System, and Heat Transport Systems shall be designed for operation at 950 °C. The NGNP shall initially operate at 850 °C with relatively short periods of operation at 950 °C until the R&D programs establish component lifetimes at 950 °C and determine the technical and programmatic risks from long-term operation at core outlet temperatures >850 °C.

4.2.3 Demonstration of High Efficiency Electricity Production

4.2.3.1 SRM Requirements

PLT 3.1.2.1 - The reactor shall have a nominal power level of 550 MW(t) with a stretch capability to about 600 MW(t).

PLT 3.1.11.1 - The NGNP shall be designed to demonstrate a capacity factor for electricity generation of ≥94% over the plant operating period from startup following a refueling to shutdown for refueling (i.e., “breaker-to-breaker”). [U/U Requirement, SRM Section 2.3.5, Fig. 1]

PLT 3.2.1 - Generate electric power using a Brayton cycle power conversion system.

PLT 3.2.2 - The overall energy conversion efficiency shall be at least 45% in the all-electric mode. Overall energy efficiency shall be as high as possible, and consistent with other key commercial parameters.

PLT 3.2.3 - Electric power production system shall be sized to produce electricity at commercial scale using 100% of the NGNP thermal energy.

PLT 3.4.2 - The NGNP shall be designed for continuous operation in either the 100% electric power production mode or in the cogeneration mode with the equivalent of up to 50 MW(t) of the reactor’s thermal energy used for hydrogen production. [Ref. 2, Section 3.4.1].

4.2.3.2 Programmatic Implementation

The NGNP design shall include a direct-cycle Power Conversion System capable of operation with a turbine inlet temperature of 950 ^{o}C. The PCS design shall be based upon the OKBM PCU design with appropriate modification for 950 ^{o}C operation (e.g., blade cooling).

4.2.4 Hydrogen Production by Multiple Technologies

4.2.4.1 SRM Requirements

PLT 3.4.1 - Hydrogen production shall be demonstrated using a thermochemical process and a high-temperature steam electrolysis (HTE) process. [Ref. 2, Section 3.4.2]

PLT 3.4.1.1 - The thermochemical process to be demonstrated by the NGNP shall be the sulfur-iodine (SI) process.

PLT 3.4.6 - The total concentration of radioactive contaminants in the hydrogen product gas and associated hydrogen production systems shall be minimized to ensure that worker and public dose limits for the integrated NGNP and hydrogen production facilities do not exceed NRC regulatory limits. [Ref. 2, Section 3.4.6].

PLT 3.4.8.1 - The hydrogen production and storage facilities shall comply with 29CFR1910.103. If the hydrogen facility produces and stores significant quantities of oxygen, compliance with 29CFR1910.104 shall also be required. [Ref. 2, Section 4.2.5]

PLT 3.4.8.2 - Emissions from the hydrogen plant shall comply with all applicable requirements of the Clean Water Act/Water Programs (CWA), 40CFR100-149, as well as compliance with all state and local requirements. [Ref. 2, Section 4.1.2]

PLT 3.4.8.3 - Emissions from the demonstration hydrogen plant shall comply with the requirements of 40CFR61, National Emissions Standards for Hazardous Air Pollutants (NESHAP), and all applicable state and local air permit requirements. [Ref. 2, Section 4.1.2]

PLT 3.4.8.4 - Exposures to any given hazardous chemical shall not exceed the maximum acceptable levels as stated in OSHA 29CFR1910.1000, Subpart Z, plus other OSHA substance-specific standards.

The purity of the product hydrogen gas shall be a minimum of {98%} (System 44).

Chemicals used in the production of hydrogen shall be recycled to the maximum extent practical to minimize the quantity of chemical waste (System 44).

Corrosion allowances for all engineering materials used in chemical processing shall be {2.95} mil/year for tubing and valves, and {19.7} mil/year for vessels and columns (System 44).

The purity of the product hydrogen gas shall be a minimum of {98%} (System 45).

The operational lifetime of the SOEC modules shall be ≥ {10 years} (System 45).

The HTE-based hydrogen production system shall utilize process heat delivered by the Secondary HTS at 800°C - 900°C…. (System 45).

4.2.4.2 Programmatic Implementation

The NGNP shall be designed with an 65 MW(t) IHX capable of operation at 950 ^{o}C which will supply process heat to both SI- and HTE hydrogen production plants. The SI and HTE hydrogen plants shall be comprised of multiple prototypical modules that can be replicated in a commercial H2-MHR based upon either technology.

4.3 References for Section 4

Hanson, D. L., “Logic for Deriving Fuel Quality Specifications,” PC-000498/0, General Atomics, March 2001.

Hanson, D. L., and J. J. Saurwein, “Development Plan for Advanced High Temperature Coated-Particle Fuels,” PC-000513, Rev. 0, General Atomics, January 2004.

Munoz, S., “Fuel Product Specification,” DOE-GT-MHR-100209, Rev. 0, General Atomics, San Diego, CA, May 1994.

[PSID] “Preliminary Safety Information Document for the Standard MHTGR,” Volumes 1-6, HTGR-84-024, Amendment B, August, 1992.

[SRM] Labar, M., D. Phelps, and J. Saurwein, “System Requirements Manual,” 911102, Rev. 0, General Atomics, March 2007.

5. DESIGN DATA NEEDS

The Design Data Needs to assure that the NGNP preconceptual design described in Section 2.2.2 meets the requirements summarized in Section 4 are presented in this section.

During the design of systems, components, and processes, the designers identify engineering development data that are needed to confirm the design (i.e., validate assumptions made in the design). In cases where this information cannot be obtained through the normally accepted level of engineering analysis, the designer prepares a Design Data Need. Each DDN defines the required data and the recommended approach to obtain the data on a schedule consistent with the program planning. The required data are obtained from development/test programs which generally fall into two basic categories: (1) technology development which provide data for upgrading design methods and validation of computer codes, and (2) component or process verification including prototypical component testing. Most of the DDNs identified during preconceptual design fall into the first category. The DDN also defines the risks associated with failure to obtain the information, along with a fall-back position which could be pursued as an alternate approach.

5.1 Methodology for Identifying Design Data Needs

The protocol used at GA for integration of design with technology development in order to maximize the benefit of the technology-development programs in terms of supporting a plant design and minimizing the technical risk of the design was introduced in Section 2.2.3 and illustrated in Figure 2-5. This section provides more detail regarding the procedures for identifying DDNs.

A systems engineering tool for developing a conceptual design to satisfy a set of top-level design requirements is functional analysis. A standard functional analysis protocol for the conceptual design of MHRs was developed for the steam-cycle MHTGR in the mid-1980s (HTGR-85-022 1985). When performing a functional analysis, the reactor designer must make certain assumptions about how a system or component will perform, especially during conceptual and preliminary design phases. In some cases, the assumption simply anticipates the expected results of a future trade study or of a more detailed analysis. In this case, the assumption is reviewed after the trade study or analysis has been completed. If the assumption is confirmed, it is replaced by the trade study, and the design is verified; if the assumption is incorrect, then the design must be modified accordingly.

In other cases, the current technology may not be sufficient to judge the correctness of the assumption at the required confidence level and this leads to a technology development need for improved technology. Conducting an R&D program typically satisfies this technology development need. Once the test program has been completed, the assumptions are reevaluated and the correctness assessed. In effect, the second type of assumption is reduced to the first type described in the preceding paragraph. This iterative procedure is repeated until all the assumptions have been eliminated through either analysis or technology development.

As an adjunct to the functional analysis protocol for the MHTGR, a formal methodology was developed for identifying technology development needs ("Design Data Needs"). The essence of the US methodology for identifying DDNs is illustrated in Figure 5-1 (DDN Procedure 1986).

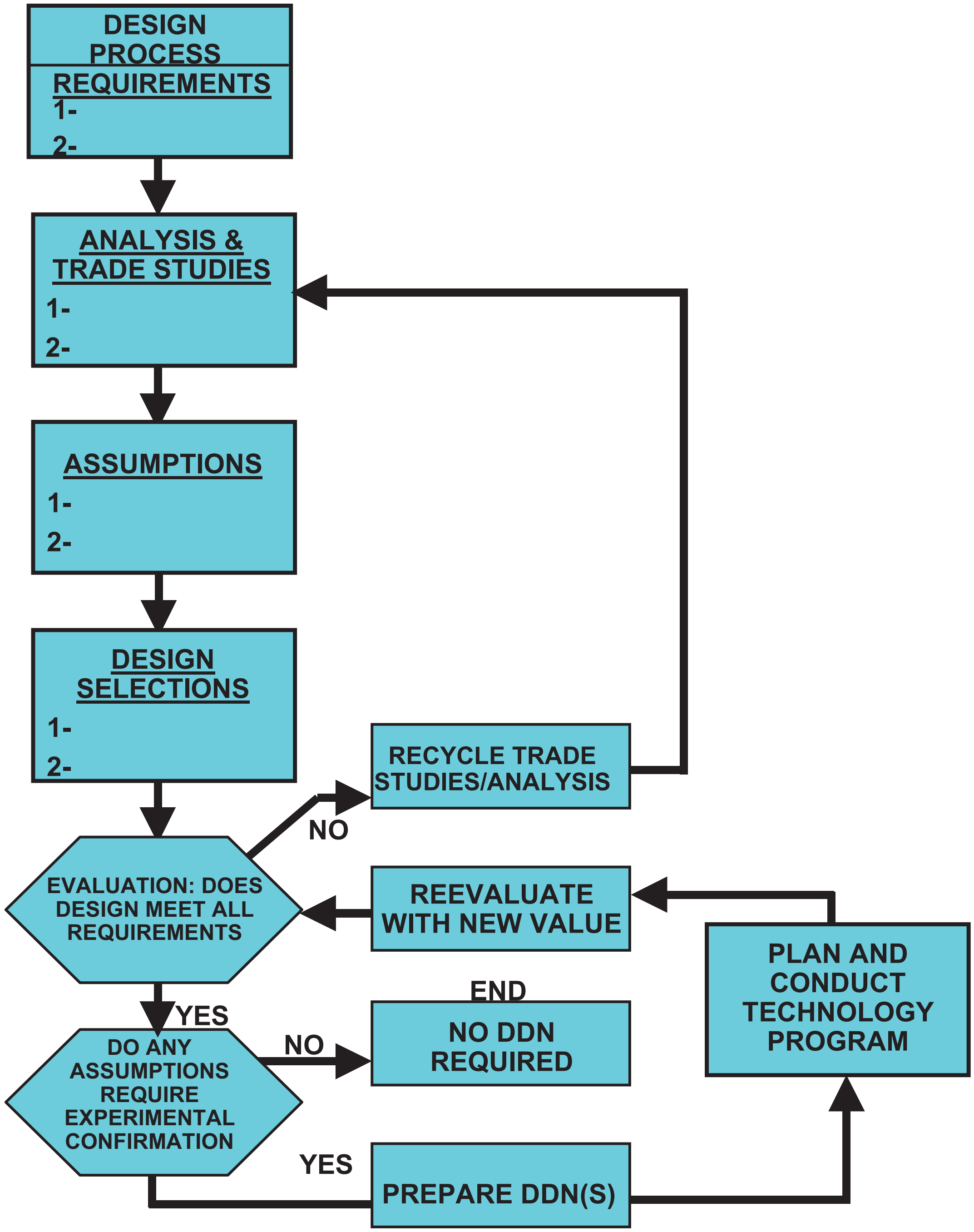

Figure 5-1. Process for Identifying DDNs

A comprehensive functional analysis has not yet been performed for the NGNP. Presumably, one will be initiated during Conceptual Design and completed during Preliminary Design. In the interim during preconceptual design, previous functional analyses performed for earlier MHR designs, augmented by engineering judgment, provided the basis for identifying a set of provisional DDNs for the NGNP that is presented below.

5.2 NGNP Design Data Needs

The Design Data Needs identified for the NGNP preconceptual design described in Section 2.2.2 are listed in Table 5-1.[19] The format used for the NGNP DDNs is the same as that used for previous DOE-sponsored MHR programs, including the NP-MHTGR and the commercial GT-MHR. An annotated DDN template is given in Appendix A.

The DDNs for various MHR designs are identified by an alpha-numeric code (e.g., N.XX.YY.ZZ) based on the definitions given in Table 5-2. This code designates the applicability to a specific reactor concept ("N"),[20] its system ("XX" following the letter), and subsystem/subgroup ("YY"). The final two numbers ("ZZ") provide sequential numbering of the DDNs within each subgroup. As shown in Table 5-2, the DDNs can have specific system applicability or multi-system applicability as appropriate. As defined in the System Requirements Manual (SRM 2007), additional systems have been added to the NGNP because of the hydrogen production plants; those new systems anticipated to generate DDNs requiring significant technology development are as follows:

13. Primary Heat Transport System (includes IHX)
42. Secondary Heat Transport System
44. SI-based Hydrogen Production System
45. HTE-based Hydrogen Production System

Many of these DDNs are generic to all MHRs, and the origin of the DDNs is also indicated in the table. In particular, most of the commercial GT-MHR DDNs (i.e., C.XX.YY.ZZ DDNs) are applicable to the NGNP (e.g., fuel and fission product DDNs, etc.). In principle, some of these commercial GT-MHR DDNs should be customized to reflect differences in the service conditions in the NGNP (e.g., higher core inlet and outlet temperatures); however, given the programmatic constraints under which this NGNP TDP was prepared, these refinements were not made. This expedient is of little or no consequence at the preconceptual design phase. It is anticipated that the NGNP TDP will be updated during the Conceptual and Preliminary Design phases, and the DDNs can be focused and refined in those updates. In any case, the actual test programs are controlled by test specifications, and it is in those specifications that test conditions need to be appropriately defined.

For those NGNP DDNs that apply to different test articles (e.g., a different alloy for the hot duct, etc.), a new DDN number ("ZZ") has been assigned, continuing the present numbering sequence for the GT-MHR DDNs in a given subsystem (i.e., N.XX.YY.ZZ). A number of NGNP DDNs, especially those relating hydrogen production, have no predecessors, and new DDNs will need to be prepared. Some of these new DDNs will be skeletal until Preliminary Design reflecting the current state of design definition; the missing information (e.g., designer's

[19] The tables for Section 5, which are generally quite large, are located at the end of the section.
[20] The letter "N" was chosen to represent NGNP; the next letter in the alphabet ("D") could have been chosen instead, but it has no mnemonic significance.

alternatives, etc.) should be added as the plant design evolves and relevant trade studies are completed.

Given the limited time and constrained budget for this task, DDNs will be developed only for those SSCs that are directly required for meeting the top-level NGNP functions defined in the SRM (2007) and/or that are judged to require significant technology development to assure that the selected designs meet the applicable requirements. Consequently, the primary emphasis will be upon critical systems and components, such as the fuel, IHX, PCU, and hydrogen plants and upon materials qualification DDNs for the fuel, graphite, and high-temperature metals that are expensive and/or of long duration. There will be less emphasis upon the balance-of-plant SSCs and on low-priority confirmatory testing needs.

As already mentioned, the DDNs for the GT-MHR (1996)[21] apply almost without exception to the NGNP preconceptual design described in Section 2.2.2 because that design includes the OKBM direct-cycle PCU which is quite similar to the earlier GA PCS design (Shenoy 1996). The DDN report for the GT-MHR (1996) included several tables which specified the tests conditions for which the experimental data were to be obtained (e.g., test articles, pressure, temperature range, etc.); these tables are reproduced here as Tables 5-3 through 5-5. These test conditions need to be updated during Conceptual Design to reflect the NGNP design (the higher core outlet temperature for the NGNP compared to the GT-MHR is an obvious example).

A number of new DDNs have been identified for the NGNP, largely because of its hydrogen production mission. These new NGNP DDNs are listed in Table 5-1 and are embellished in Table 5-6; the latter table contains the following DDN elements: (1) DDN number, (2) DDN title, (3) Data Needed, (4) Design Significance, and (5) Fallback Position and Consequences of Nonexecution. This degree of DDN definition is judged sufficient and appropriate for preconceptual design. During Conceptual Design, these new DDNs should be completed per the template given in Appendix A, and periodically updated as the design matures and feedback is obtained from regulators and potential customers.

5.2.1 Fuel/Fission Products

The fuel/fission product DDNs identified for the commercial GT-MHR (DDNs 07.XX.YY in Table 5-1) appear to be largely adequate for the NGNP at the preconceptual design with several caveats discussed below.

The fuel process DDNs appear to require no modifications at this stage. This conclusion would change if the reference fuel design was modified (e.g., for a single particle design which is not recommended in the PCDSR) or if the fuel in AGR-1 performs differently than expected, indicating a need for further optimization of process conditions.

With regard to fuel performance, DDNs for 10%-enriched TRISO UO_2 (produced by NFI) will need to be added if that fuel is used for the initial NGNP core (and perhaps for early reloads as

[21] (GT-MHR DDNs 1996), which is 518 pages long, is readily available to NGNP Project participants; consequently, it is not reproduced here.

well). At minimum, DDNs for fuel performance under irradiation conditions and under core heatup conditions can be anticipated. The exact nature of these NFI UO_2 DDNs will be likely determined by negotiations with the NRC. The test conditions for the GT-MHR DDNs may also need to be extended at a later time to reflect higher fuel temperatures for 950 °C operation if core optimization efforts are unsuccessful.

The radionuclide transport DDNs appear adequate for NGNP preconceptual design with several exceptions. DDNs for radionuclide transport in the VLPC need to be added (Hanson 2007). In addition, the GT-MHR DDNs for RN transport in the primary circuit may need to be modified to reflect the selection of different metal alloys (e.g., IN 617 for the IHX) and higher metal service temperatures in the NGNP.

5.2.2 Reactor System Materials

The Reactor System materials DDNs for the commercial GT-MHR appear adequate for NGNP preconceptual design with several exceptions.

5.2.2.1 Core Graphites

The generic graphite DDNs for the commercial GT-MHR appear adequate for the NGNP with the exceptions that a replacement graphite for H-451 must be identified and qualified and the service temperature range extended.

5.2.2.2 Ceramics

The ceramic materials DDNs for the commercial GT-MHR appear adequate for NGNP.

5.2.2.3 High-Temperature Metals

The reactor internals DDNs for the commercial GT-MHR appear adequate for NGNP with the exception that the service temperature range needs extended. The DDNs for the Primary Heat Transport System, including the IHX, are addressed in Section 5.2.3.1.

5.2.2.4 Reactor Pressure Vessel Materials

Two new DDNs have been identified for the NGNP RPV constructed of 2¼Cr-1Mo, one for vessel surface emissivity and another one for Nil-Ductility Transition Temperature (NDTT) shift data. The 9Cr-1Mo-V RPV DDNs for the commercial GT-MHR should be retained since 9Cr is the recommended backup RPV material.

5.2.3 Energy Transfer Technology

The DDNs for the Primary Heat Transport System and Secondary Heat Transport System are summarized below.

5.2.3.1 Primary Heat Transport System

The key component of the Primary Heat Transport System which has significant technology development needs is the Intermediate Heat Exchanger. The preferred choice for the IHX design is the printed circuit heat exchanger. The PCHE technology achieves high effectiveness and low LMTD in a compact heat exchanger with reasonable pressure drops across the exchanger. PCHEs consist of alternating metallic plates in which microchannels have been chemically etched and then joined together under high pressure and temperatures to form a diffusion-bonded heat transfer core. PCHE technology has been applied to numerous industries but has yet to be applied in the nuclear industry – especially for gas-cooled reactors at the very high temperatures.

Additional DDNs are associated with the Primary Helium Circulator (PHC) which is designed to use magnetic bearings to support the shaft of the circulator.

5.2.3.2 Secondary Heat Transport System

For the Secondary Heat Transport System, the significant DDNs are associated with the high temperature isolation valves required on the hot leg of the system. Additional DDNS are associated with internal insulation. The design of the secondary circulators are based on the same technology proposed for the Primary Helium Circulator so that additional DDNs are not required.

5.2.4 Power Conversion System

The PCS DDNs are organized as the baseline DDNs for qualifying the OKBM PCU design for operation at 850 ^{o}C and the incremental DDNs for extending operation to 950 ^{o}C.

5.2.4.1 PCS Operation at 850 ^{o}C

It was recognized from the beginning that the vertical integrated PCS concept poses several technical challenges with respect to individual component design and their arrangement within a single PCS vessel. On the other hand, it was also clear that there are substantial technical and economic incentives for such a selection. While the higher efficiency and lower cost of the integrated concept were the major drivers, minimizing the volume and area of the highest temperature regions of the machine, plus the reduced volume and area of the primary pressure boundary as a whole were seen as additional technical advantages.

Given the technical challenges associated with the integrated PCS configuration, the PCS design development was carefully monitored by the GT-MHR project through a series of design reviews, both by internal experts and by independent third party experts. The results of these technical reviews were thoroughly reviewed and evaluated to identify the uncertainties and unconfirmed assumptions (i.e., technical issues) in the science or engineering on which the design is based. A series of Design Data Needs were then prepared to define the data needed to resolve these uncertainties and unconfirmed assumptions, and technology demonstration plan (2005) was prepared to describe the overall PCS development program.

The DDNs for the reference vertical integrated PCS design represent the input requirements for the detailed development and testing activities identified within the PCS Technology Demonstration Plan (PCU TDP 2005) for the US/RF International GT-MHR. These DDNs are identified and defined in the lower-tier technology demonstration plans for the systems, subsystems, and components that were developed for the PCU preliminary design. Table 5-7 (Table 14 from PCU TDP 2005) provides a summary of these DDNs, the documents in which they are described, and a summary of the planned means of data acquisition.

From inspection of Table 5-7, it is evident that the DDN format used by the US/RF International GT-MHR program is somewhat different than the format used for the various US MHR programs (Appendix A) although the functional differences are not large. When an umbrella NGNP TDP is prepared during Conceptual Design, it may be programmatically desirable to repackage the PCU DDNs in the standard US MHR (Appendix A) format.

5.2.4.2 PCS Operation at 950 ^{o}C

Several upgrades to the PCS will be required for operation with an inlet temperature of 950 ^{o}C, thus generating some new DDNs. In particular, turbine blade temperatures must be kept below the creep limit in order to achieve the desired lifetime of 60,000 hours. Consequently, blade cooling will likely be required which was not the case for 850 ^{o}C. In addition to blade cooling, a thermal barrier coating will likely need to be applied to the blades. The stability of such coating in a high temperature He environment will need to be confirmed experimentally. The hot gas duct (which connects reactor and PCS) will be to be better insulated in order to reduce heat losses and protect its structural materials. In particular, quality and thickness of the high-temperature insulation will have to be upgraded. There also needs to be additional data to support the recuperator design which would validate its performance and longevity at higher temperatures relative to the reference GT-MHR case. Design data needs for PCS operation at 950 ^{o}C are outlined in Table 5-6.

5.2.5 Design Verification and Support

A significant number of DV&S DDNs have been identified for the commercial GT-MHR, especially for the Reactor System (see Table 5-1). The DDNs related to RS materials are described separately in Sections 5.2.2 since they are largely generic. Most of these GT-MHR DV&S DDNs are expected to apply to the NGNP design because the nuclear heat source is quite similar with the exception of higher core outlet temperature with the latter. As the NGNP design matures, especially with the selection of lower-level components, additional DV&S DDNs can be anticipated.

5.2.6 Hydrogen Production

The DDNs identified for the SI- and HTE hydrogen production processes are summarized below. Both technologies are immature; consequently, it is highly probable that additional DDNs related to process scale-up and integration will be identified as the designs evolve.

5.2.6.1 Sulfur-Iodine Thermochemical Water-Splitting

A basic requirement for equipment design in the chemical process industries is a robust understanding and definition of the chemical thermodynamics involved. Vapor-liquid equilibrium data for distillation column design, reaction kinetics for reactor design, and liquid mixture heat capacity data for heat exchangers are examples of the types of data that are necessary for design. Expected corrosion rates are also necessary for vessel and piping designs. No new unit operations must be developed to support a thermochemical process like the SI cycle, but the physical property data to be used with mature design methods must be reliable.

Four categories of DDNs exist to support the collection of more data in each of the process sections. Their goal is to reduce uncertainties (and the attendant design margins) in equipment design for the SI process. The DDN categories are: (1) sulfuric acid decomposition; (2) Bunsen reaction; (3) HI decomposition; and (4) materials compatibility. These DDNs are described in Tables 5.1, 5.2, and 5.6. Process control work is currently going on at laboratory scale, and will continue at larger scales. Materials testing is ongoing, but manufacturing techniques for large-scale items constructed from non-metallic materials should be studied. The decomposition of HI to hydrogen and iodine is a key step in the process. Not only does it produce the product hydrogen, but also the method selected for decomposition can play a significant role in the overall process cost and efficiency. Liquid decomposition of HI could be explored further as an efficient avenue for high-pressure hydrogen production. Operation of the integrated lab-scale device for closed-loop operation will also be a useful tool for satisfying the DDNs required for process scale-up.

5.2.6.2 High Temperature Electrolysis

The enabling technology for HTE is based on solid oxide fuel cells. Solid-oxide electrolyzer (SOE) concepts, based on both planar-cell and tubular-cell technologies, are currently being developed. SOE technology based on the planar-cell concept is being developed as part of the DOE Nuclear Hydrogen Initiative and involves collaboration between INL and Ceramatec of Salt Lake City, UT. A potential issue for the planar-cell concept is stack durability and sealing as the result of thermal cycling. Tubular cells have less active cell area per unit volume than planar cells but are less susceptible to this issue. Toshiba Corporation is currently developing an SOE concept based on tubular-cell technology.

The GA team believes that both the planar-cell and tubular-cell technologies are promising concepts for future commercialization and recommends that both concepts be developed at least through the pilot-scale demonstration stage so that tradeoffs between capital costs and long-term performance can be accurately characterized.

The DDNs for both HTE concepts are essentially the same, but separate technology programs will have to be conducted because of the configuration differences. As indicated in Table 5-1, there are four categories of DDNs: (1) basic SOE cell design and performance, (2) design of SOE units that include multiple cells, (3) design of SOE modules that include multiple SOE units, and (4) other equipment associated with the HTE plant (steam generators and other heat exchangers). Design data required for instrumentation and control will be obtained as part of DDNs for the pilot- and engineering-scale programs.

5.2.7 Spent Fuel Disposal

The preferred option for HTGR spent fuel disposition has been shown to be the direct disposal of unprocessed spent fuel elements in a geologic repository (Lotts 1992, Richards 1994, Richards 2002). In fact, unprocessed HTGR spent fuel elements are a nearly ideal waste form for permanent geologic disposal; the ceramic coated-particle fuel, encapsulated in nuclear graphite blocks, represents a far smaller proliferation risk and a far more robust radionuclide containment system than Zircaloy-clad, commercial LWR spent fuel.

In the process of performing the cited assessments of coated-particle fuel performance and radionuclide transport in a repository environment, it became apparent that certain additional experimental data, primarily related to the long-term stability of coated-particle fuel, would serve to increase the confidence in the predictions, and a confirmatory test and analysis plan defining experimental programs to generate such data was prepared for the commercial GT-MHR (Hanson 2002).

The spent fuel disposal DDNs for the commercial GT-MHR (Hanson 2002) are listed in Table 5-1; the corresponding DDNs for the NGNP should be virtually identical.

5.3 DDN Schedule Requirements

As indicated on the DDN template (Appendix A), each DDN defines the schedule for when the data must be available to support major programmatic milestones (e.g., the start of final design, FSAR submittal, etc.). Rather than list the required delivery date for each individual DDN in Table 5-1, it was judged more practical at the preconceptual design stage to address the schedule requirements by DDN category. The results are summarized in Table 5-8 in which the delivery dates are tied to major NGNP programmatic milestones; a nominal calendar date consistent with the Option 2 schedule given in the PMPP (2006) is also provided.

Some of the delivery dates given in Table 5-8 are significantly later in the design process than called in the commercial GT-MHR DDNs (2006). For example, component DV&S data are generally called for a year prior to completion of Final Design. For the GT-MHR, some of these DV&S DDNs call for data at the end of Preliminary Design (i.e., two years earlier). These DDNs were delayed with the rationale that the design of a first-of-a-kind plant would have to progress beyond a two-year Preliminary Design before the SSCs could be sufficiently well defined to prepare the test specifications for a DV&S test program. The need dates for each NGNP DDN should be carefully evaluated during Conceptual Design.

5.4 References for Section 5

[DDN Procedure] “DOE Projects Division Program Directive #16: HTGR PROGRAMS - Design Data Needs (DDNs) Interim Procedure,” PD#16, Rev. 1, February 1986.

[GT-MHR DDNs] “600 MW(t) Gas Turbine Modular Helium Reactor Design Data Needs,” DOE-GT-MHR-100217, General Atomics, July 1996.

Hanson, D. L., and M. B. Richards, “[Commercial GT-MHR] Spent Fuel Disposal Confirmatory Test and Analysis Plan,” PC-000503, Rev. 0, General Atomics, June 2002.

Hanson, D. L., and J. M. Bolin, “Radionuclide Transport in a Vented Low-Pressure Containment,” PC-000541, Rev. 0, General Atomics, April 2007.

HTGR-85-022, “Procedures and Guidelines for Functional Analysis,” General Atomics, June 1985.

Lotts, A. L., et al., “Options for Treating High-Temperature Gas-Cooled Reactor Fuel for Repository Disposal,” ORNL/TM-12017, Oak Ridge National Laboratory, February 1992.

[PCU TDP] PCU “Technology Demonstration Program Plan, Revision 2005 (Draft),” Product No: 08.03-006.01A, OKBM, 2005 *(Business Confidential).*

Shenoy, A., “Gas Turbine-Modular Helium Reactor (GT-MHR) Conceptual Design Description Report,” GA Document 910720, Rev. 1, General Atomics, July 1996.

[PPMP] Weaver, K., et al., “Next Generation Nuclear Plant Project Preliminary Project Management Plan,” INL/EXT-05-00952, Rev. 1, Idaho National Laboratory, March 2006.

Richards, M. B. and D. W. Ketchen, "PC-MHR Spent Fuel Disposal: Preliminary Evaluation of Whole-Element Disposal Using Multipurpose Canisters," GA/DOE-164-94, General Atomics, September 1994.

Richards, M., “Assessment of GT-MHR Spent Fuel Characteristics and Repository Performance,” GA Document PC-000502, Rev. 0, General Atomics, March 2002.

[SRM] Labar, M., D. Phelps, and J. Saurwein, “System Requirements Manual,” 911102, Rev. 0, General Atomics, March 2007.

Table 5-1. NGNP Design Data Needs

DDN NO.	DDN TITLE	SOURCE
C.07.00	FUEL/FISSION PRODUCT	
C.07.01	Fuel Fabrication	Commercial GT-MHR
C.07.01.01	UCO Kernel Process Development	“
C.07.01.02	Fuel Particle Coating Process Development	“
C.07.01.03	Fuel Compact Fabrication Process	“
C.07.01.04	Quality Control Test Techniques Development	“
C.07.01.05	Fuel Product Recovery Development	“
N.07.01.06	Mass Production of High Quality UCO TRISO Fuel	New
N.07.01.07	As-manufactured Quality of LEU UO_2 (NFI extended burnup fuel)	New
C.07.02	Fuel Performance	Commercial GT-MHR
C.07.02.01	Coating Material Property Data	“
C.07.02.02	Defective Particle Performance Data	“
C.07.02.03	Thermochemical Performance Data for Fuel	“
C.07.02.04	Fuel Compact Thermophysical Properties	“
C.07.02.05	Normal Operation Fuel Performance Validation Data	“
C.07.02.06	Accident Fuel Performance Validation Data	“
C.07.02.07	Fuel Proof Test Data	“
N.07.02.08	Irradiation Performance of LEU UO_2 (NFI extended burnup fuel))	New
N.07.02.09	Accident Performance of LEU UO_2 (NFI extended burnup fuel))	New
C.07.03	Radionuclide Transport	Commercial GT-MHR
C.07.03.0I	Fission Gas Release from Core Materials	“
C.07.03.02	Fission Metal Effective Diffusivités in Fuel Kernels	“
C.07.03.03	Fission Product Effective Diffusivities in Particle Coating	“
C.07.03.04	Fission Product Diffusivities/Sorptivities in Graphite	“
C.07.03.05	Tritium Permeation in Heat Exchanger Tubes	[Later]
C.07.03.06	Tritium Transport in Core Materials	Commercial GT-MHR
C.07.03.07	Radionuclide Deposition Characteristics of Structural Materials	“
C.07.03.08	Decontamination Protocols for Turbine Alloys	“
C.07.03.09	Radionuclide Reentrainment Characteristics for Dry Depressurization	“
C.07.03.10	Radionuclide Removal Characteristics for Wet	Commercial GT-MHR

DDN NO.	DDN TITLE	SOURCE
	Depressurization	
C.07.03.11	Characterization of the Effects of Dust on Radionuclide Transport	“
C.07.03.12	Fission Product Transport in a Vented Low-Pressure Containment	“
C.07.03.13	Decontamination Efficiency of Pressure Relief Train Filter	“
C.07.03.14	Fission Gas Release Validation Data	“
C.07.03.15	Fission Metal Release Validation Data	“
C.07.03.16	Plateout Distribution Validation Data	“
C.07.03.17	Radionuclide "Liftoff" Validation Data	“
C.07.03.18	Radionuclide "Washoff" Validation Data	“
N.07.03.19	Physical and Chemical Forms of RNs Released during Core Heatup	PC-000541
N.07.03.20	RN Sorptivities of VLPC Surfaces	“
N.07.03.21	Qualification of Coatings with High Iodine Sorptivity	“
N.07.03.22	Validation Data for Predicting RN Transport in VLPC	“
C.07.04	<u>Core Corrosion Data</u>	Commercial GT-MHR
C.07.04.01	Coated B_4C Corrosion Data	“
C.07.04.02	Core Matrix Materials Corrosion Data	“
C.07.04.03	Core Corrosion Methods Validation Data	“
N.07.05	<u>Spent Fuel Disposal</u>	GT-MHR Spent Fuel Disposal Plan
N.07.05.01	Long-Term Mechanical Integrity of Stressed TRISO Coatings	“
N.07.05.02	PyC Coating Oxidation Rates (Air)	“
N.07.05.03	SiC Coating Oxidation Rates (Air)	“
N.07.05.04	Matrix Oxidation Rates (Air)	“
N.07.05.05	H-451 Graphite Oxidation Rates (Air)	“
N.07.05.06	Graphite Noncombustibility Demonstration	“
N.07.05.07	PyC Coating Corrosion Rates (Groundwater)	“
N.07.05.08	SiC Coating Corrosion Rates (Groundwater)	“
N.07.05.09	Matrix Corrosion Rates (Groundwater)	“
N.07.05.10	H-451 Graphite Corrosion Rates (Groundwater)	“
N.07.05.11	Radionuclide Leaching Rates From UCO Kernels	“
N.07.05.12	C-14 Content Of Matrix And Graphite	“
N.07.05.13	Chemical Impurities In H-451 Graphite	“
N.07.05.14	Radionuclide Leaching Rates From Graphite	“

DDN NO.	DDN TITLE	SOURCE
C.11.00	REACTOR System (RS)	
C.11.01	Neutron Control System	Commercial GT-MHR
C.11.01.01	Control Rod Instrumentation and Control Verification	“
C.11.01.02	Qualification of Electromechanical Components of Neutron Control Assembly	“
C.11.01.03	Control Rod Drive Design Verification	“
C.11.01.04	Reserve Shutdown Control Equipment Design Verification	“
C.11.01.05	In-Core Flux Mapping Unit (IFMU) Drive Design Verification	“
C.11.01.06	In-Core Flux Mapping Unit (IFMU) Design Verification	“
C.11.01.07	Neutron Control Assembly Flow and Leak Design Verification	“
C.11.01.08	Guide Tubes Flow Induced Vibration Design Verification	“
C.11.01.09	Demonstration of Remote Handling and Maintenance of Neutron Control Assembly	“
C.11.01.10	Neutron Control Assembly Seismic Qualification	“
C.11.01.11	Neutron Control Assembly Qualification Test	“
C.11.02	Reactor Internals and Hot Duct	Commercial GT-MHR
C.11.02.01	Core Support Strength Data	“
C.11.02.02	Hot Duct Integrity Verification	“
N.11.02.10	Effects of Primary He and Temperature on Metallic Reactor Internals Materials	New
N.11.02.11	Irradiation Effects on Metallic Reactor Internals Materials	Significant Changes
N.11.02.12	Irradiation Effects on Hot Duct Metals	“
N.11.02.13	Effects of Primary He and Temperature on Hot Duct Materials	“
N.11.02.14	Fibrous Insulation Material Properties	“
C.11.02.15	Hard Ceramic Insulation Properties Data	Commercial GT-MHR
N.11.02.16	Emissivity of Metallic Materials Reactor Internals (MRI)	Significant Changes
C.11.03	Reactor Core	Commercial GT-MHR
C.11.03.01	Core Column Vibration Data	“
C.11.03.02	Control Rod Vibration	“
C.11.03.03	Core Element Dynamic Strength Data	“
C.11.03.04	Core Element Failure Mode Data	“
C.11.03.05	Control Rod Shock Absorber Data	“
C.11.03.06	Control Rod Structural Integrity Data	“
C.11.03.11	Graphite Multiaxial Strength Data	“

DDN NO.	DDN TITLE	SOURCE
C.11.03.12	Graphite Fatigue Data	“
C.11.03.13	Graphite Mechanical Properties Data	“
C.11.03.14	Graphite Irradiation Induced Dimensional Change Data	“
C.11.03.15	Graphite Irradiation Induced Creep Data	“
C.11.03.16	Graphite Thermal Properties Data	“
C.11.03.17	Graphite Fracture Mechanics Data	“
C.11.03.18	Graphite Corrosion Data	“
C.11.03.19	Graphite Corrosion Data for Methods Validation	“
C.11.03.20	Graphite Destructive and Nondestructive Examination Data	“
C.11.03.21	Graphite Coke Source Qualification	“
C.11.03.22	Reserve Shutdown Pellet Process Development Data	“
C.11.03.23	Graphite Oxidation Data for Postulated Accidents	“
C.11.03.24	Properties of High Temperature Control Rod Materials	“
C.11.03.31	Verification of Neutron Detectors and Cabling	“
C.11.03.41	Fuel Element Channel Flow Data	“
C.11.03.42	Control Rod Channel Flow Data	“
C.11.03.43	Bottom Reflector/Core Support Pressure Drop and Flow Mixing Data	“
C.1I.03.44	Metallic Plenum Element and Top Reflector Pressure Drop and Flow Distribution	“
C.11.03.45	Core Crossflow Test Data	“
C.11.03.46	Core Fluctuation Test Data	“
C.11.03.51	Integral Nuclear Data Measurement at Temperature for GT-MHR Physics Methods Validation	“
C.11.03.52	Critical Experimental Data for GT-MHR Physics Methods Validation	“
C.11.04	<u>Reactor Service Equipment</u>	Commercial GT-MHR
C.11.04.01	Reactor Equipment Service Facility Tools Design Verification	“
C.11.04.02	Reserve Shutdown Vacuum Tool Design Verification	“
C.11.04.03	Neutron Detector Service Equipment Design Verification	“
C.11.04.04	Metallic Reactor Internals (MRI) 151 and Surveillance Equipment Design Verification	“
C.11.04.05	Metallic Core Support ISI and Surveillance Equipment Design Verification	“
C.11.04.06	Graphite Reactor Internals (GRI) and Core Support ISI and Surveillance Design Verification	“
C.12.00	VESSEL SYSTEM (VS)	

DDN NO.	DDN TITLE	SOURCE
C.12.01	Vessels	Commercial GT MHR
N.12.01.01	Irradiation Data for Reactor Vessel Materials (modified 9Cr-IMo, SA-387 Grade 91, Class 2 Plate and SA-336 Grade F91 Forging)	Significant Changes
N.12.01.02	Properties of Heavy Section Vessel Materials (SA-387 Grade 91, Class 2 Plate/SA-336 Grade F91 Forging) at Elevated Temperatures	Significant Changes
N.12.01.03	Reactor Vessel Emissivity (Modified 9Cr-IMo, ABB-CE SA387 Grade 91, Class 2 Plate and SA-336 Grade F91 Forging)	Significant Changes
C.12.01.04	Helium Seal Data for Bolted Closures	“
N.12.01.05	Irradiation Data for Reactor Vessel Material, (2¼ Cr – 1Mo)	New
N.12.01.06	Reactor Pressure Vessel Emissivity (2¼ Cr – 1Mo)	New
N.13.00	PRIMARY HEAT TRANSPORT SYSTEM (PHTS)	
N.13.01	Primary Helium Circulator (PHC)	Toshiba
N.13.01.01	Effects of Primary Coolant Helium and Temperature on Primary Heat Transport System Circulator Materials	New
N.13.01.02	Circulator Magnetic and Catcher Bearings Design Verification	“
N.13.01.03	Circulator Prototype Design Verification	“
N.13.02	Intermediate Heat Exchanger (IHX)	Toshiba
N.13.02.01	Effects of Helium and Temperature on IHX Materials	New
N.13.02.02	Confirmation of PCHE Core Strength	“
N.13.02.03	Confirmation of Design Evaluation Procedure	“
N.13.02.04	Confirmation of PCHE Core Temperature Distribution	“
N.13.02.05	Confirmation of IHX Thermal Hydraulic Characteristics	“
N.13.02.06	IHX Acoustic Test	“
N.13.02.07	IHX Insulation Verification Tests	“
N.13.02.08	IHX Seal Tests	“
N.13.02.09	IHX Flow Induced Vibration Test	“
		“
C.14.00	SHUTDOWN COOLING SYSTEM (SCS)	
C.14.01	Shutdown Circulator	Commercial GT-MHR
C.14.01.01	SCS Circulator Magnetic and Catcher Bearings Design Verification	“
C.14.01.02	SCS Circulator Prototype Impeller Aerodynamic and Acoustic Test Data	“

DDN NO.	DDN TITLE	SOURCE
C.14.01.03	SCS Circulator Prototype Test in High Pressure Test Facility (HPTF)	“
C.14.01.04	Shutdown Circulator Loop Shut-off Valve (SLSV) Life Cycle Test Data	“
N.14.01.05	Irradiation Effects on SCS Circulator Materials	New
N.14.01.06	Effects of Primary Coolant Helium and Temperature on SCS Circulator Materials	New
C.14.04	Shutdown Heat Exchanger (SHE)	Commercial GT-MHR
C.14.04.01	SHE Insulation Verification Tests	“
C.14.04.02	SHE Vibrational Fretting Wear and Sliding Wear of TRDs for Bare Tubes	“
C.14.04.03	SHE Instrumentation Attachment Test	“
C.14.04.04	SHE Bare Tubes Inspection Methods and Equipment	“
C.14.04.05	SHE Shroud Seal Test	“
C.14.04.06	Acoustical Response of the SHE Helical Bare Tube Bundle	“
C.14.04.07	SHE Inlet Flow and Temperature Distribution Test	“
C.14.04.08	SHE Tube Bundle Local Heat Transfer and Flow Resistance Characteristics	“
C.14.04.09	SHE Tube Helical Coil Program	“
C.14.04.10	SHE Lead-in/Lead-out Expansion Loop Tube Design and Fabrication	“
C.14.04.11	Irradiation Effects on Shutdown Cooling System Heat Exchanger Materials	New
C.14.04.12	Effects of Primary Coolant Helium and Temperature on Shutdown Cooling System Heat Exchanger Materials	New
C.16.00	REACTOR CAVITY COOLING SYSTEM (RCCS)	Commercial GT-MHR
C.16.00.01	Emissivity of RCCS Panel Metal Surfaces	“
C.16.00.02	Wind Tunnel Test of RCCS I/O Structure	“
C.16.00.03	Integrated RCCS Performance	“
C.16.00.04	RCCS Cooling Panel Heat Transfer Coefficient and Friction Factor	“
C.16.00.05	Effective Conductivity of Core Blocks	“
C.16.00.06	Buoyancy Induced Fluid Mixing in a High Aspect Ratio Cavity	“
C.21.00	FUEL HANDLING AND STORAGE SYSTEM (FH&SS)	
C.21.01	Core Refueling	Commercial GT-MHR
C.21.01.01	Fuel Handling Machine/Handling Mechanism Design Verification	“

DDN NO.	DDN TITLE	SOURCE
C.21.01.02	Fuel Transfer Cask Component Design Verification	“
C.21.01.03	Element Hoist and Grapple Assembly Robot Design Verification	“
C.21.0I.04	Verify Fuel Handling System Instrumentation and Control	“
C.21.01.05	Integrated Fuel Handling System Test Data	“
C.21.01.06	Fuel Handling Equipment Positioner Design Verification	“
C.21.01.07	Fuel Handling Equipment Support Structure Design Verification	“
C.21.01.08	Fuel Sealing and Inspection Equipment Design Verification	“
C.21.01.09	Inflatable Seal and SN Identification Tests Design Verification	[Later]
C.31.00	REACTOR PROTECTION SYSTEM (RPS)	
C.31.01	Safety Protection and Instrumentation	Commercial GT-MHR
C.31.01.01	Verify Helium Mass Flow Measurement Instrumentation	“
C.31.01.02	Verify Conduction Cooldown Temperature Monitoring Instrumentation	[Later]
C.34.00	PLANT CONTROL, DATA AND INSTRUMENTATION SYSTEM (PCD&IS)	
C.34.01	Nuclear Island Control and Instrumentation	Commercial GT-MHR
C.34.01.01	Verify Core Inlet and Outlet Helium Temperature Measurement Instrumentation	[Later]
C.34.01.02	Verify Plateout Probe Operation	[Later]
N.41.00	POWER CONVERSION SYSTEM (PCS)	PCU TDP (Table 5-7)
N.41.01	Turbomachine	
N.41.01.01	Service Lifetime of Uncooled Turbine Blades at 950 ^{o}C	New
N.41.01.02	Verification of Blade Cooling Design	New
N.41.01.03	Thermal Barrier Coatings for Turbine Blades	New
N.41.02	Recuperator	
N.41.02.01	Creep Data for Candidate Recuperator Metals	New
N.41.02.02	Recuperator Support System. Design Verification	New
N.41.04	Ducts and Seals	
N.41.04.01	PCS Duct Design Verification & Support	New

DDN NO.	DDN TITLE	SOURCE
N.42.00	SECONDARY HEAT TRANSPORT SYSTEM (SHTS)	
N.42.01	SHTS Circulator	Toshiba
	(Secondary Circulator DDNs are covered by Primary Helium Circulator DDNS due to design similarity)	
N.42.02	Isolation Valves	
N.42.02.01	Effects of Primary Coolant Helium and Temperature on SHTS Piping and Valve Materials	New
N.42.02.02	High Temperature Isolation Valve Prototype Design Verification	“
N.44.00	SI-BASED H_2 PRODUCTION SYSTEM (SI)	GA
N.44.01	Sulfuric Acid Decomposition	
N.44.01.01	Catalyst Performance	New
N.44.02	Bunsen Reaction	
N.44.02.01	Bunsen Reaction Physical Chemistry	New
N.44.02.02	Refined Thermodynamic Model	“
N.44.03	Hydrogen Iodide Decomposition	
N.44.03.01	HI/H_2 Membrane Separation	New
N.44.03.02	Refined Thermodynamic Model	“
N.44.03.03	Liquid HI Decomposition	“
N.44.04	Materials Compatibility	
N.44.04.01	Corrosion performance	New
N.44.04.02	Equipment Manufacturability	“
N.45.00	HTE-BASED H_2 PRODUCTION SYSTEM (HTE)	Toshiba
N.45.01	SOE Cells	
N.45.01.01	Electrode/Electrolyte Materials	New
N.45.01.02	SOEC Design and Performance	“
N.45.02	SOE Units	
N.45.02.01	SOE Unit Design and Performance	New
N.45.02.02	SOE Multi-Unit Integration and Performance	“

DDN NO.	DDN TITLE	SOURCE
N.45.03	SOE Modules	
N.45.03.01	SOE Pilot-Scale Module Demonstration	New
N.45.03.02	SOE Engineering Scale Module Demonstration	“
N.45.03.03	NGNP SOE Multi-Module Demonstration	“
N.45.04	HTE Plant Supporting Equipment	
N.45.04.01	HTE Steam Generator/Superheater	New
N.45.04.02	HTE Heat Exchangers	“

Table 5-2. NGNP DDN Identification Code Protocol

MHR DESIGN DESIGNATOR	
A.	DDNs maintained during preliminary and final design of 350 MW(t) MHTGR.
B.	DDNs maintained during preliminary and final design of 450 MW(t) MHTGR.
C.	DDNs developed during the conceptual design of 600 MW(t) GT-MHR.
N.	DDNs developed during preconceptual design of NGNP
Multi-Systems Applicability	
01.00	Plant Performance
02.00	Availability and Maintenance
03.00	In-Service Inspection (ISI)
04.00	Plant Dynamics
05.00	Safety and Reliability
06.00	Plant Seismic
07.00	Fuel/Fission Product
07.01	Fuel Fabrication
07.02	Fuel Performance
07.03	Radionuclide Transport
07.04	Core Corrosion Data
07.05	Spent Fuel Disposal
08.00	Decay Heat Removal
Specific System Applicability	
11.00	Reactor System
11.01	Neutron Control
11.02	Reactor Internals & Hot Duct
11.03	Reactor Core
11.04	Reactor Service Equipment
12.00	Vessel System

MHR DESIGN DESIGNATOR	
12.01	Vessels
12.02	Vessel Support
12.03	Vessel Pressure Relief
13.00	Primary Heat Transport System[22] (added for NGNP)[23]
13.01	Primary Helium Circulator (PHC)
13.02	Intermediate Heat Exchanger (IHX)
14.00	Shutdown Cooling System
14.01	Shutdown Circulator
14.02	Shutdown Cooling Heat Removal Control
14.03	Shutdown Cooling System Service Equipment
14.04	Shutdown Heat Exchanger
16.00	Reactor Cavity Cooling System
21.00	Fuel Handling and Storage System
21.01	Core Refueling
31.00	Reactor Protection System
31.01	Safety Protection and Instrumentation
34.00	Plant Control, Data and Instrumentation System
34.01	Nuclear Island Control and Instrumentation
41.00	Power Conversion System[24]
41.01	Turbomachine
41.02	Recuperator
41.03	Precooler/Intercooler
41.04	Ducts and Seals
41.05	Power Conversion System Service Equipment
42.00	Secondary Heat Transport System (added for NGNP)

[22] IHX DDNs will use this code; subsystems to be added as required. Steam generator DDNs for steam-cycle MHRs carry this identification code as well.
[23] Additional codes for other systems to be defined as necessary using system numbers from the SRM.
[24] DDNs for the OKBM PCU design modified for 950 ^{o}C operation shall be identified "N.41.YY.ZZ."

MHR DESIGN DESIGNATOR	
42.01	SHTS Circulator
42.02	Isolation Valves
43.00	Secondary Helium Purification System (added for NGNP) [25]
44.00	SI-based Hydrogen Production System (added for NGNP)
44.01	Sulfuric Acid Decomposition
44.02	Bunsen Reaction
44.03	Hydrogen Iodide Decomposition
44.04	Materials Compatibility
45.00	HTE-based Hydrogen Production System (added for NGNP)
45.01	SOE Cells
45.02	SOE Units
45.03	SOE Modules
45.04	HTE Plant Supporting Equipment

[25] There may be no new DDNs for this system, depending upon the chosen design.

Table 5-3. Reactor Service Conditions for Normal Operation[26]

Parameter	Value
Environment	Helium
Nominal fuel operating temperature range (instantaneous @ full power) Additional allowance for design uncertainties	[550 to 1250 ^oC] [150 ^oC]
Maximum nominal time-averaged fuel temperature Additional allowance for design uncertainties	[1150 ^oC] [70 ^oC]
Maximum nominal fissile particle burnup Design allowance on fissile burnup	[25%] FIMA [1%] FIMA
Maximum nominal fertile particle burnup Design allowance on fertile particle burnup	[6%] FIMA [0.5%] FIMA
Maximum fast neutron fluence (E >29 fJ) Design allowance on fast neutron fluence	[5 x 10^{25} n/m^2] [0.5 x 1025 n/m^2]
Maximum coolant pressure	[7.07 MPa]
Range of coolant impurity levels during power operation: H_20 CO Total oxidants H_2 CH_4	 [0. 07 to 0.7 Pa] (0 01 to 0.1 ppmv)] [1.5 to 6 Pa (0. 2 to 0. 8 ppmv)] [<7 Pa (<1 ppmv)] [3 to 10 Pa (0.5 to 1.5 ppmv)] [0.3 to 1.5 Pa (0.05 to 0.2 ppmv)]
Nominal fuel temperature range, refueling	[100 to 500 ^oC]
Environment, refueling	Helium @ [0.1 MPa]

[26] Commercial GT-MHR with 850 ^oC core outlet temperature.

Table 5-4. Reactor Conditions for Core Conduction Cooldown Events[27]

Parameter	Value
Environment for transient events (depressurized conduction cooldown)	He; He/H_20/CO/H_2; He/CO/N_2
H_20	[Negligible to TBD]
CO	[0 to 35 kPa (0 to 0.35 atm)]
N_2	[0 to 65 kPa (0 to 0.65 atm)]
Range of H_20 impurity levels (pressurized conduction cooldown)	[1 to TBD kPa (0.01 to TBD atm)]
Fuel temperature - depressurized conduction cooldown	[550 to 1600 ^{o}C]
Allowance in peak for design uncertainty	[125 ^{o}C]
Fuel temperature - pressurized conduction cooldown	[550 to 1300 ^{o}C]
Allowance in peak for design uncertainty	[100 ^{o}C]
Duration of event:	
Depressurized conduction cooldown	[150 hr]
Pressurized conduction cooldown	[100 hr]

[27] Commercial GT MHR with 850 ^{o}C core outlet temperature.

Table 5-5. PCS and PV Conditions during Normal Operation and Accidents[28]

Parameter	Value
Normal Operation	
Reynolds Number	> 5000
PCS materials (candidate materials)	[IN 100, SS316L, ½%Cr-½%Mo, 9%Cr-1%Mo-V][29]
Metal temperature range: IN 100 (turbine) SS316L (recuperator) ½%Cr-½%Mo, (precooler/intercooler) 9% Cr-1%Mo-V (vessel)	 [450 to 900 ^oC] [100 to 550 ^oC] [100 to 150 ^oC] [100 to 550 ^oC]
Particulate matter: Composition Particle size distribution Gasborne concentration Surface loading	 [Amorphous carbon, ferritic metal oxide, graphite] [0.01 to 10 μm] [3 x 10^{-3} g/m^3] [5 g/m^2]
Rapid Depressurization	
Environment	He
Coolant outlet temperature range	[TBD to 850 ^oC]
Range of coolant impurity levels during power operation: H_20 CO CO_2 Total oxidants H_2	 [14 to TBD Pa] [35 Pa] [14 Pa] [<70 to TBD Pa] [70 Pa]
Coolant pressure range	70 to 1 atm
Shear ratio[30]	[0.5 - 3]
Blowdown duration	[1 to 2 min]
Reynolds Number	TBD
Metal temperature range: IN 100 (turbine) SS316L (recuperator) ½%Cr-½%Mo (precooler/intercooler) 9% Cr-1%Mo-V (vessel)[23]	 [450 to TBD ^oC] [100 to TBD ^oC] [100 to TBD ^oC] [100 to TBD ^oC]

[28] Commercial GT MHR with 850 ^oC core outlet temperature.
[29] 9%Cr-1%Mo-V was the leading candidate PV material for the commercial GT-MHR; 2¼%Cr-1%Mo or SA 508 is the recommended PV material for the NGNP since it is doubtful that Grade 91 can be qualified in time to support the NGNP schedule.
[30] Shear ratio = ratio of wall shear stress during depressurization to that during normal operation.

Parameter	Value
Water Ingress (PCS)	
Environment	He/$H_2$0
Coolant temperature range	[100 to TBD ^oC]
Range of coolant impurity levels	[0.01 to TBD atm $H_2$0]
Coolant pressure range	70 to 1 atm
Metal temperature range: IN 100 (turbine) SS316L (recuperator) ½%Cr-½%Mo (precooler/intercooler) 9% Cr-1%Mo-V (vessel)[23]	 [450 to TBD ^oC] [100 to TBD ^oC] [100 to TBD ^oC] [100 to TBD ^oC]
Reynolds Number	> 5000
Shear ratio	<1.0
Steam quality	[0 to 100%]
Contact time	[0.1 to TBD hr]

Table 5-6. New NGNP DDNs

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
N.07.00 Fuel/Fission Products				
N.07.01	**Fuel Fabrication**			
N.07.01.06	Mass Production of High Quality UCO TRISO Fuel	Demonstration that UCO TRISO fuel meeting NGNP fuel product and process specifications can be economically mass produced by construction and operation of an integrated fuel fabrication pilot plant capable of supplying NGNP reload fuel segments on the required schedule.	UCO TRISO fuel meeting NGNP fuel product specifications has not been mass produced. There is no domestic fuel supplier of TRISO fuel.	Rely upon previous US fuel manufacturing experience for the design of fuel fabrication facility and introduce process improvements during startup. High risk that process yields will be low and that fuel production costs will be excessively high, leading to unfavorable fuel cycle cost projections for future commercial VHTRs.
N.07.01.07	As-manufactured Quality of LEU UO_2 (NFI extended burnup fuel)	As-manufactured fuel quality data to demonstrate that NFI is capable of manufacturing fuel to NGNP fuel quality requirements.	NFI must demonstrate that it can mass produce fuel to NGNP fuel product specification.	Attempt to license the NGNP without these data using a license by test approach (unlikely to be accepted by the NRC and involves high risk) or obtain fuel from another source (no known candidates). A delay in licensing will delay startup of the NGNP
N.07.02	**Fuel Materials**			
N.07.02.08	Irradiation Performance of LEU UO_2 (NFI extended burnup fuel)	Irradiation performance data for proof test fuel fabricated by NFI to demonstrate that NFI extended burnup fuel meets NGNP in-pile fuel performance requirements	The irradiation performance database for the NFI extended burnup fuel is limited and is judged by GA to be inadequate to support NGNP licensing	Attempt to license the NGNP without this data using a license by test approach (unlikely to be accepted by the NRC and involves high risk) or obtain fuel from another source (no known candidates). A delay in licensing will delay startup of the NGNP

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
N.07.02.09	Accident Performance of LEU UO_2 (NFI extended burnup fuel)	Accident condition performance data for proof test fuel fabricated by NFI to demonstrate that NFI extended burnup fuel meets NGNP accident condition fuel performance requirements	The accident condition fuel performance database for the NFI extended burnup fuel is limited and is judged by GA to be inadequate to support NGNP licensing	Same as above
N.07.03	**Radionuclide Transport**			
N.07.03.19	Physical and Chemical Forms of Radionuclides Released during Core Heatup	The physical form and chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) released from the core into the primary circuit and VLPC during heatup accidents must be determined experimentally with particular attention to the effects of coolant chemistry on composition. Reliance on thermochemical analysis is not sufficient.	The transport behavior of key RNs in (I, Sr, Cs, Te, and Ag) in the VLPC cannot be determined without knowing their physical form and chemical composition. Iodine is the highest priority.	Assume that all RNs are released from core in elemental form and as atomic vapor. NRC may not give credit for RN holdup in primary circuit and VLPC requiring tighter limits on core release and fuel failure.
N.07.03.20	RN Sorptivities of VLPC Surfaces	Data are needed to characterize the deposition of I, Cs, Ag and Te on prominent VLPC surfaces (paint, concrete, etc.), including the sorptivities of these nuclides as a function of temperature, partial pressure, surface state, and coolant chemistry for normal operating conditions. Particular attention should be given to the effects of dust (DDN C.07.03.11).	Molecular deposition is expected to be the dominant removal mechanism for key RNs, including iodines, in the VLPC.	Assume that VLPC surfaces are perfect sinks for volatile RNs at the relatively low temperatures expected. NRC may not give credit for RN holdup in primary circuit and VLPC requiring tighter limits on core release and fuel failure.
N.07.03.21	Qualification of Coatings with High Iodine Sorptivity	The potential for increasing iodine retention in the VLPC during core heatup accidents by application of highly sorptive coatings along dominant flow pathways should be investigated by a literature survey. The effectiveness of leading candidates needs to be confirmed experimentally for MHR-specific service conditions.	Molecular deposition is expected to be the dominant removal mechanism for key RNs, including iodines, in the VLPC. Highly sorptive coatings or paints may increase iodine retention.	Measure sorptivities of uncoated VLPC surface materials. Opportunity to enhance plateout in VLPC lost. Consequences unknown; sorptivities of uncoated surfaces may be adequate.
N.07.03.22	Validation Data for Predicting RN Transport in VLPC	Integral test data are needed to independently validate the methods describing the transport behavior of condensable RNs in the VLPC under dry and wet core conduction cooldown conditions. The effects of temperature,	LWR test experience (e.g., the PHEBUS tests) has demonstrated that a representative source of radionuclides (i.e., irradiated	Unless safety analysis methods validated under MHR-specific conditions, NRC may not give credit for RN holdup in VLPC requiring

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
		coolant chemistry, surface state, and aerosols must be treated explicitly. The chemical composition of the key radionuclides (I, Sr, Cs, Te, and Ag) must also be determined with particular attention to the effects of coolant chemistry on composition (compound formation). Integral test data are needed where the reactor core, primary circuit and VLPC are properly represented and coupled	fuel) is essential for reliable containment response tests. The irradiated fuel must contain a sufficient quantity of I-131.	tighter limits on core release and fuel failure. (Use of LWR integral data for code validation not credible.)
N.11.00 Reactor Core				
N.11.02	**<u>Reactor Internals and Hot Duct</u>**			
N.11.02.10	Effects of Primary He and Temperature on Metallic Reactor Internals Materials	Data are needed on the effects of elevated temperature corrosion from primary coolant helium impurities during normal operation and accident conditions on the following properties of metallic reactor internals Alloy 800H base metal and weldments. See DDN C.11.02.13 for properties list and DDN modifications.	Interaction with primary coolant helium impurities and long-term thermal aging can seriously degrade materials properties, e.g., strength and ductility . These data are required to determine the extent of possible degradation in order to use proper property values in design.	Develop a more conservative design and/or utilize higher strength and/or primary He-compatible materials.
N.11.02.11	Irradiation Effects on Metallic Reactor Internals Materials	Add creep crack growth to DDN C.11.02.11 and modify expected irradiation and temperature conditions per preconceptual design.	Irradiation at elevated temperatures (above 600 ^{o}C) reduces ductility of candidate material (Alloy 800H), and makes materials more vulnerable to excessive creep crack growth. These data are necessary, in concert with other data, for possible codification purposes.	Develop more conservative design and/or utilize higher strength materials.
N.11.02.12	Irradiation Effects on Metallic Hot Duct Materials	Add creep crack growth to DDN C.11.02.12 and modify expected irradiation and temperature conditions to include conditions for thermal barrier canisters or plates for	Irradiation at elevated temperatures (above 600 ^{o}C) reduces ductility of candidate material (Alloy 800H), and	Develop more conservative design and/or utilize higher strength materials

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
		insulation attachments inside hot duct.	makes materials more vulnerable to excessive creep crack growth. These data are necessary, in concert with other data, for possible codification purposes.	
N.11.02.13	Effects of Primary He and Temperature on Hot Duct Materials	Add creep crack growth to DDN C.11.02.13 and modify expected irradiation and temperature conditions to include conditions for thermal barrier canisters or plates for insulation attachments inside hot duct.	Irradiation at elevated temperatures (above 600 °C) reduces ductility of candidate material (Alloy 800H), and makes materials more vulnerable to excessive creep crack growth. These data are necessary, in concert with other data, for possible codification purposes	Develop more conservative design and/or utilize higher strength materials
N.11.02.14	Fibrous Insulation Material Properties	Add physical properties (thermal conductivity and heat capacity); long-term thermal, dimensional, and compositional stability; high temperature strength; resistance to pressure drop, vibration, and acoustic loads; radiation and primary coolant helium corrosion resistance; stability to dust and gas release; and thermal creep) to DDN C.11.02.14.	Data are needed to provide confidence in and support design adequacy.	Employ and active cooling system to eliminate need for passive thermal protection. Utilize metallic materials with greater high-temperature strength for metallic components for which insulation is proposed so that insulation will not required.
N.11.02.16	Emissivity of Metallic Reactor Internals Materials	Add control rod components and modify temperatures to reflect expected conditions per preconceptual design to DDN C.11.02.16	Data are needed for performing proper heat transfer analysis and determine component operating temperatures.	Design by conservatively estimating emissivity values based on data for other materials.
N.12.00 Vessel System				
N.12.01	**<u>Vessels</u>**			
N.12.01.05	Irradiation Data for Reactor Vessel Material, (2¼ Cr–1Mo)	Data are needed to characterize the neutron-induced changes in fracture toughness, tensile strength, and creep properties for the reactor vessel plate and forging materials, and weldments at temperatures and neutron flux,	High confidence in RPV integrity is essential for design certification.	Extrapolate existing irradiation effects data to higher temperatures. May become a licensing issue.

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
		fluence, and spectrum levels expected for the NGNP.		
N.12.01.06	Reactor Pressure Vessel Emissivity (2¼ Cr–1Mo)	Data are needed to demonstrate that the thermal emissivity of the vessel materials is at least 0.8 for temperatures that occur during conduction cooldown events over the entire life of the vessel. In addition, the emissivities for normal operation and upset conditions are needed in order to characterize the vessel temperatures and heat loss.	Influences peak fuel temperatures during core heatup accidents.	Extrapolate existing emissivity data to higher temperatures. NRC may require conservatism which will increase predicted fuel temperatures.
N.13.00 Primary Heat Transport System				
N.13.01	**PHTS Circulator**			
N.13.01.01	Effects of Primary He and Temperature on Primary Heat Transport Circulator Materials	Data are needed on the effects of elevated temperature corrosion from primary coolant helium impurities during normal operation and accident conditions on the following properties of metallic primary heat transport circulator candidate material base metal and weldments. See DDN C.11.02.13 for properties list and DDN modifications.	Interaction with primary He impurities and long-term thermal aging can seriously degrade materials properties, e.g., strength and ductility . This data are required to determine the extent of possible degradation in order to use proper property values in design.	Rely upon existing database and add design margin to compensate for uncertainties. Increased risk of equipment failure.
N.13.01.02	Circulator Magnetic and Catcher Bearings Design Verification	Data are required to establish: (1) static and dynamic axial thrust load capacities, stiffness and damping coefficients of radial bearings for the entire operating speed range, (2) sensitivity of electronic control system to outside disturbances, (3) rotor dynamic response to externally induced unbalance loads, and (4) useful life of catcher bearings.	Required for use of magnetic bearing technology.	Use oil-lubricated bearings which have the potential for oil ingress into the primary coolant.
N.13.01.03	Circulator Prototype Design Verification	Data on the functional capability of the entire circulator system including motor/control/-circulator compatibility are required. Data include: (1) aerodynamic performance of the inlet, loop shutoff valve, circulator impeller and diffuser; (2) motor thrust bearing performance, (3) overspeed capability, (4) structural	Prototype testing required prior to release of hardware production drawings.	Relay on subassembly tests or demonstrate performance after installation in vessel during hot flow test. Increases risk of design not meeting performance requirements.

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
		integrity of rotating parts and supports, (5) noise levels and frequencies, (6) vibration characteristics and critical speeds, (7) shutdown and hot restart capability including hot soak.		
N.13.02	**Intermediate Heat Exchanger (IHX)**			
N.13.02.01	Effects of Primary Helium and Temperature on IHX Materials	Data are needed on the effects of elevated temperature corrosion from primary coolant helium impurities during normal operation and accident conditions on the material properties of metallic IHX candidate material base metal and weldments. See DDN C.11.02.13 for properties list and DDN modifications.	Interaction with primary coolant impurities and long-term thermal aging can seriously degrade materials properties, e.g., strength and ductility. These data are required to determine the extent of possible degradation in order to use proper property values in design.	Develop a more conservative design and/or utilize higher strength and/or primary coolant helium-compatible materials.
N.13.02.02	Confirmation of the PCHE Core Strength	The PCHE core is the product of a layered structure of thin plates that are joined together using a diffusion bonding procedure. It is necessary to verify the strength of these diffusion bonded joints.	Required for use of PCHE technology	Adopt a helical coil IHX which may require an increased LMTD to in order to keep the size reasonable.
N.13.02.03	Confirmation of Design Evaluation Procedure	ASME Section III Subsection NH is planned to be applied to the design of the PCHE core. It is necessary to confirm whether the result of the design has enough margin to pressure- and heat-resistance in the PCHE core by testing. Also, it is necessary to confirm the inspectability of the PCHE core.	Required for use of PCHE technology	Adopt a helical coil IHX which may require an increased LMTD to in order to keep the size reasonable.
N.13.02.04	Confirmation of PCHE Core Temperature Distribution	As applied to the IHX, it is necessary to confirm by experiment the temperature distribution accompanied by analytical evaluation.	Required for use of PCHE technology	Adopt a helical coil IHX which may require an increased LMTD to in order to keep the size reasonable.
N.13.02.05	Confirmation of IHX Thermal Hydraulic Characteristics	It is necessary to confirm by experiment the flow distribution throughout the IHX (both primary and secondary inlets and outlets) accompanied by analytical evaluation.	Required for use of PCHE technology	Adopt a helical coil IHX which may require an increased LMTD to in order to keep the size reasonable.

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
N.13.02.06	IHX Acoustic Test	Data are needed to produce frequency spectra and sound pressure levels that may be generated by the IHX as a function of flow velocities.	Required to reduce risk associated with PCHE technology.	Rely on analysis and avoid large resonant plates or other acoustically sensitive structures in the design.
N.13.02.07	IHX Insulation Verification Tests	Physical and operational characteristics of insulation are required relative to thermal cycling, mechanical and acoustic vibrations, and effects of flow and thermal gradients.	Required to assure insulation meets performance requirements over its design lifetime.	Rely on manufacturer's data.
N.13.02.08	IHX Seal Tests	Various sliding seals, expansion joints, and other seals joining surfaces are expected in the IHX design for installation and replacement purposes. Data are needed to confirm the design feasibility, measure leak rates under operating conditions, and measure the influence of various factors on seal performance.	Confirmatory testing required before completion of Preliminary Design to minimize changes during Final Design.	Redesign to eliminate as many seals as possible. Increases risk of design not meeting performance requirements.
N.13.02.09	IHX Flow Induced Vibration Test	Data are needed to accurately determine the flow-induced vibration characteristics around the IHX and its associated piping. The flow-induced excitation mechanisms of concern are turbulent buffeting, vortex shedding and fluid elastic instability.	Needed to prevent the IHX design from being overly conservative which would result in higher cost.	Designer will have to use the most conservative correlations if this design data need is not satisfied.
N.14.00 Shutdown Cooling System				
N.14.01	**SCS Circulator**			
N.14.01.05	Irradiation Effects on SCS Circulator Materials	Data are needed on the effects of neutron irradiation during normal operation and accident conditions on the physical properties of metallic SCSC material base metal and weldments.	Neutron irradiation at elevated temperatures can seriously degrade materials properties, e.g., strength and ductility . These data are required to determine the extent of possible degradation in order to use proper property values in design.	Develop a more conservative design by using shielding or reducing allowables and/or utilizing higher strength materials.
N.14.01.06	Effects of Primary He and Temperature on SCS	Data are needed on the effects of primary helium impurities during normal operation and	Interaction with primary coolant impurities and long-	Develop a more conservative design and/or utilize higher

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
	Circulator Materials	accident conditions on the physical properties of metallic SCSC material base metal and weldments.	term thermal aging can seriously degrade materials properties, e.g., strength and ductility . These data are required to determine the extent of possible degradation in order to use proper property values in design.	strength and/or primary coolant helium impurities-compatible materials.
N.14.04	**<u>SCS Heat Exchanger</u>**			
N.14.04.11	Irradiation Effects on Shutdown Cooling System Heat Exchanger (SCSHX) Materials	Data are needed on the effects of neutron irradiation during normal operation and accident conditions on the physical properties of metallic SCSHX material base metal and weldments.	Neutron irradiation at elevated temperatures can seriously degrade materials properties, e.g., strength and ductility . These data are required to determine the extent of possible degradation in order to use proper property values in design.	Develop a more conservative design by using shielding or reducing allowables and/or utilizing higher strength materials.
N.14.04.12	Effects of Primary He and Temperature on Shutdown Cooling System Heat Exchanger Materials	Data are needed on the effects of primary coolant impurities during normal operation and accident conditions on the physical properties of metallic SCSCX material base metal and weldments.	Interaction with primary coolant impurities and long-term thermal aging can seriously degrade materials properties, e.g., strength and ductility . These data are required to determine the extent of possible degradation in order to use proper property values in design.	Develop a more conservative design and/or utilize higher strength and/or primary coolant helium impurities-compatible materials.
N.41.00 Power Conversion System: see Table 5-7 (PCU TDP 2005) for 850 ^{o}C Baseline DDNs; see below for 950 ^{o}C operation.				
N.41.01	**<u>Turbomachine</u>**			
N.41.01.01	Service Lifetime of Uncooled Turbine Blades at 950 ^{o}C	Determine service lifetimes of IN 100 and IN 738 as function of temperature and stress over range of 850 – 950 ^{o}C.	Maximum electrical generation efficiency if turbine can operate at >850 ^{o}C without blade cooling for acceptable lifetime.	Accept reduced service lifetime or add blade cooling and thermal barrier coatings. Higher cost electricity costs

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
				and reduced availability.
N.41.01.02	Verification of Blade Cooling Design	Turbine blade cooling design, including amount of cooling required, heat transfer coefficient, pressure drops, effectiveness of cooling on maintaining the blade temperature below 850 ^{o}C, needs to be verified experimentally.	The turbine blade materials do not need cooling to achieve desired life of 60,000 hours at 850 ^{o}C. The lifetime of turbine blades will be reduced at T >850 ^{o}C.	Design turbine to minimize blade stresses and accept a shorter service lifetime for the turbine. Higher cost electricity costs and reduced availability.
N.41.01.03	Thermal Barrier Coatings for Turbine Blades	Thermal barrier coatings will be needed on the first few stages to reduce temperature to <850 ^{o}C. Identify optimal coating composition (low Co content) and application technique. Confirm coating stability and effectiveness in high temperature, high velocity He for service lifetime of 60,000 hr.	The life turbine blades will be significantly reduced if maximum temperature exceeds 850 ^{o}C. If coating spalls off prematurely, blades may overheat and fail. Spallation fragments may be transported into core and become activated, generating another radiation source in the primary circuit.	Use high temperature materials for turbocompressor (TC) blades. Design turbine with blade cooling only and accept a shorter service lifetime for the turbine. Higher cost electricity costs and reduced availability.
N.41.02	**Recuperator**			
N.41.02.01	Creep Data for Candidate Recuperator Metals	Obtain creep data on thin recuperator plates at higher temperatures (500 – 600 ^{o}C).for leading candidate and backup recuperator metals (e.g., SS 316).	Increase in inlet temperature to the TC will increase the outlet temperature from TC and inlet temperature to the recuperator from current design value of 510 ^{o}C to ~ 580 ^{o}C. The creep life of the recuperator material will need to be determined.	Use high temperature materials for recuperator. Higher cost or reduced service life.
N.41.02.02	Recuperator Support System. Design Verification & Support	Verify experimentally acceptable stresses in the support system of the recuperator at higher temperature differences.	Redesign the support system for higher thermal stresses associated with higher temperature differences.	Use higher strength metals for recuperator support system. Higher cost or reduced service life.
N.41.04	**Ducts and Seals**			
N.41.04.01	PCS Duct Design Verification & Support	Perform mock-up tests to confirm that internal ducting and other mechanical connections	These are used to minimize the leakages from different	Rely upon good design practice.

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
		within the PCS meet design requirements (e.g., leak tightness).	parts of the vessel to each other.	Increased risk of internal leaks within PCS.
N.XX.00 Design Verification and Support – Reactor System (11), Vessel System (12), etc.				
		No new DV&S DDNs for nuclear source systems identified at this writing, but new DDNs should be anticipated during Conceptual and Preliminary Designs		
N.42.00 Secondary Heat Transport System				
N.42.02	**Isolation Valves**			
N.42.02.01	High Temperature Isolation Valve Prototype Design Verification	Data are needed to assess performance of valve internal insulation, valve seat material, seal performance and structural integrity.	Isolation valves are a required and their development is a high priority.	An active internal cooling system could improve valve performance but at a much greater cost and complexity than simple insulation.
N.44.00 Sulfur-Iodine Thermochemical Water-Splitting				
N.44.01	**Sulfuric Acid Decomposition**			
N.44.01.01	Catalyst Performance	Catalyst activity and lifespan as a function of temperature and system pressure	Better understanding of actual plant capacity factors and capital/operating costs	Use demonstration plant to determine catalyst performance. More expensive at larger scales
N.44.02	**Bunsen Reaction**			
N.44.02.01	Bunsen Reaction Physical Chemistry	Reaction kinetics and enthalpies as a function of temperature and system pressure.	Optimized reactor design, reduced cost, reduced side reaction	Use current data with associated uncertainties. Potentially overdesigned reactor, leading to increased costs. Oversized volume could promote side reactions
N.44.02.02	Refined Thermodynamic Model	Compilation of chemical and phase equilibrium data from University-based work	Improved design reliability	Use current models with associated uncertainties. The requirement of increased flexibility in the equipment could lead to increased capital costs

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
N.44.03	**Hydrogen Iodide Decomposition**			
N.44.03.01	HI/H_2 Membrane Separation	- H_2 permeability - HI/H_2 separation factor - Membrane life	Reduced cost compared to refrigerated phase separation	Use baseline refrigerated phase separation. No additional cost savings in this area.
N.44.03.02	Refined Thermodynamic Model	$H_2O/HI/H_2/I_2/PdI_2$ vapor-liquid-liquid equilibrium data	Improved design reliability	Using current models with associated uncertainties. The requirement of increased flexibility in the equipment could lead to increased capital costs
N.44.03.03	Liquid HI Decomposition	- Catalyst recovery techniques and efficiency of recovery - Kinetics	- Higher process efficiency - Lower capital cost - High pressure (50 bar) H_2	Current design with gas-phase decomposition. Baseline efficiency and costs
N.44.04	**Materials Compatibility**			
N.44.04.01	Corrosion performance	Corrosion rates for desired engineering materials subjected to typical manufacturing stresses (bends, welds, etc.)	Optimal design, reduced cost	Large corrosion allowances, leading to higher costs and decreased efficiencies.
N.44.04.02	Equipment Manufacturability	Fabrication techniques for non-metallics	Optimal design, reduced cost	Use of metallic (tantalum) heat exchangers, with potentially higher costs and decreased efficiencies.
N.45.00 High Temperature Electrolysis (Toshiba Design)				
N.45.01	**SOE Cells**			
N.45.01.01	Electrode / Electrolyte Materials	Basic data on ionic conductivity, ohmic loss, material stability at high temperature, structural properties, corrosion resistance, and thermal properties.	Data needed to support SOE cell design and model SOE cell performance	Adopt materials and data from NHI planar-cell technology program. Materials may not be optimum for tubular cells.
N.45.01.02	SOEC Design and Performance	Lifetime testing (50,000 h) of individual cells. Hydrogen production rate as a function of time and temperature.	Data needed to support SOE unit design and model SOE unit performance	Adopt the planar-cell technology. Places reliance on single HTE technology.
N.45.02	**SOE Units**			

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
N.45.02.01	SOE Unit Design and Performance	Lifetime testing of individual (1 Nm^3/h) units. Hydrogen production rate as a function of time and temperature.	Data needed to support SOE pilot-scale module design and model SOE pilot-scale module performance	Adopt the planar-cell technology. Places reliance on single HTE technology.
N.45.02.02	SOE Multi-Unit Integration and Performance	Performance of multiple (~10) integrated units. Evaluation of manifolding configurations and flow rates. Evaluation of electrical configurations	Data needed to support SOE pilot-scale module design and model SOE pilot-scale module performance	Adopt the planar-cell technology. Places reliance on single HTE technology.
N.45.03	**SOE Modules**			
N.45.03.01	SOE Pilot-Scale Module Demonstration	Long-term performance of a pilot-scale module (60 Nm^3/h) at high pressure. Procedures for startup, control, and maintenance. Assessment of instrumentation. Data for flowsheet assessment and validation, including steam generation, hydrogen recycle and recuperation of heat from product gases.	Data needed to support SOE engineering-scale module design and model SOE engineering-scale module performance	Adopt the planar-cell technology. Places reliance on single HTE technology
N.45.03.02	SOE Engineering Scale Module Demonstration	Long-term performance of an engineering-scale module (600 Nm^3/h) at high pressure. Procedures for startup, control, and maintenance. Assessment of instrumentation. Data for flowsheet assessment and validation, including steam generation, hydrogen recycle and recuperation of heat from product gases.	Initial demonstration of a prototype module for commercialization	Adopt the planar-cell technology. Places reliance on single HTE technology
N.45.03.03	NGNP SOE Multi-Module Demonstration	Long-term performance of an integrated HTE plant consisting of 10 engineering-scale modules (600 Nm^3/h) at high pressure. Procedures for startup, control, and maintenance of multiple modules. Assessment of instrumentation. Data for flowsheet assessment and validation, including steam generation, hydrogen recycle and recuperation of heat from product gases.	Provides basis for commercialization of nuclear hydrogen production using HTE technology.	Adopt the planar-cell technology. Places reliance on single HTE technology
N.45.04	**HTE Plant Supporting Equipment**			
N.45.04.01	HTE Steam Generator/ Superheater	To be determined after design is more fully developed and materials are selected.	Needed to support HTE plant design.	[TBD] determined after design is more fully developed and materials are selected.

DDN No.	DDN Title	Data Needed	Data Significance	Fallback Position & Consequences
N.45.04.02	HTE Heat Exchangers	To be determined after design is more fully developed and materials are selected.	Needed to support HTE plant design.	[TBD] after design is more fully developed and materials are selected.

Table 5-7. Design Data Needs for OKBM Power Conversion Unit[31]

System or Component	TDP		DDN		DDN Acquisition Means	
	Product	Title	Number	Need	ID	Summary
6 PCU	2.3-33C, Rev.1	PCU TDP	DDN-6-1	Data on mounting, repair, maintenance and replacement of PCU components	T-6-1	Creation of full-scale mockups of PCU congestion with PCU cavity elements and tests with mockups of process tooling to repair, mount, maintain and replace PCU components
			DDN-6-2	Data on computer simulation to validate PCU assemblability	T-6-2	Procurement of required computer programs and hardware and check of PCU assemblability using computer model
6.1 PCU IVM	2.3-31C	PCU IVM TDP	DDN-6.1-1	Data on helium flows mixing in low pressure collector header	T-6.1-1	Creation and testing of the model of helium circulation path from precooler module outlet to LPC inlet in air
			DDN-6.1-2	Data on helium flows mixing in the collector header at HPC inlet	T-6.1-2	Creation and testing of the model of helium circulation path from intercooler module outlets to HPC inlet in air
			DDN-6.1-3	Data on low pressure helium flow rate distribution among recuperator modules	T-6.1-3	Creation and testing of the model of helium circulation path from turbine outlet to recuperator modules outlet in air
			DDN-6.1-4	Data on operability of PCU IVM large-size compensator operating at high temperature and pressure drops	T-6.1-4	Creation and testing of full-scale compensator model under conditions maximally approximated to reactor operation conditions at the test facility
6.2 TM CPS Mixer	02.03-43A	TM CPS Mixer TDP	DDN-6.2-1	Data on mixing helium flows in the area from turbine outlet chamber to recuperator support inlet	T-6.2-1	Creation and tests of a model for helium circulation path from mixer distribution chamber inlet and from turbine outlet to recuperator support inlet

[31] Table 14 from PCU TDP (2005). When an umbrella NGNP TDP is prepared during Conceptual Design, it may be programmatically desirable to repackage these DDNs in the standard US MHR DDN format (Appendix A).

System or Component	TDP		DDN		DDN Acquisition Means	
	Product	Title	Number	Need	ID	Summary
10 TM	02.03-34B, Rev.0	TM TDP, Including Cost and Schedule	DDN-10-1	Operating properties and characteristics of TM, including indices of reliability, safety, mounting and dismounting	T-10-1	Creation of pilot TM sample and its testing at "cold" test facility
					T-10-1.1	TM testing in PCU during RP equipment start and adjustment work
10.1 EMB	2.3-115A, Rev.1	EMB TDP	DDN-10.1-1	Data on stability of vertical rotor rotation at full electromagnetic suspension	T-10.1-1	Creation of Minimockup and testing equipment. Study of rotor rotation parameters at full electromagnetic suspension
			DDN-10.1-2	Data on operability of elements in the "power amplifier-electromagnet-rotor-sensor-controller" system	T-10.1-2	Creation of EMB model and test facility. Study of EMB operability and characteristics
			DDN-10.1-3	Data on schematic-design solutions of rotor position sensors and electronic boards for sensors signal processing	T-10.1-3	Creation of EMB sensor models and test facility. Study of operability and characteristics of EMB sensor models
			DDN-10.1-4	Data on operability of EMB together with CS within multi-support electromagnetic suspension of TM rotor	T-10.1-4	Creation of scaled model of multi-support electromagnetic suspension of TM rotor and test facility. Study of scaled model
10.1 EMB	2.3-115A, Rev.1	EMB TDP	DDN-10.1-5	Characteristics of EMB together with CS and operating properties	T-10.1-5	Creation of EMB prototypes and test equipment. Tests of EMB prototypes
			DDN-10.1-6	Operating properties and characteristics of EMB system of TM rotor, including reliability, safety and mounting indices	T-10.1-6	Test of EMB system for full-scale TC
					T-10.1-6.1	Test of EMB system for TM rotor at "cold" test facility
					T-10.1-6.2	Test of EMB system for TM rotor during acceptance tests of TM in PCU

System or Component	TDP		DDN		DDN Acquisition Means	
	Product	Title	Number	Need	ID	Summary
10.2 CB	2.3-115B, Rev.1	CB TDP	DDN-10.2-1	Data on service parameters of friction pair materials and solid lubricants for CBs	T-10.2-1	Creation of friction pair samples ant their preliminary tests at L-1129 test facility in helium environment
			DDN-10.2-2	Data on operability of CBs	T-10.2-2	Creation and testing of CB mockups
					T-10.2-2.1	Creation and testing of CB test specimens in helium
					T-10.2-2.2	Creation and testing of CBs as a part of pilot TC sample
11 Generator	2.3-112A, Rev.1	Generator TDP	DDN-11-1	Verified design procedures for vertical generators with helium cooling and magnetic bearings	T-11-1	Creation and testing of generator mockup
			DDN-11-2	Design of generator stator winding insulation	T-11-2	Creation and testing of insulation specimens of generator stator winding components
			DDN-11-3	Parameters of exciter winding insulation	T-11-3	Creation and testing of EMB and exciter winding insulation specimens
			DDN-11-4	Parameters of generator electric leading	T-11-4	Creation and testing of EMB and generator electric leading experimental specimens
12 TC	2.3-114A, Rev.1	TC TDP	DDN-12-1	Data on operability and effectiveness of TC compressors	T-12-1	Creation and tests of scaled LPC of the TC
			DDN-12-2	Data on operability of diaphragm coupling	T-12-2	Creation and tests of scaled diaphragm coupling
			DDN-12-3	Data on operability and effectiveness of brush and labyrinth seals of TC rotor	T-12-3	Creation and tests of experimental brush and labyrinth seals of TC rotor

System or Component	TDP		DDN		DDN Acquisition Means	
	Product	Title	Number	Need	ID	Summary
12 TC	2.3-114A, Rev.1	TC TDP	DDN-12-4	Materials for high-temperature TC elements	T-12-4	Search, selection and tests of structural materials for long-term static and long-term cyclic strength for operation conditions: lifetime not less than 60000h, temperature 850C and specified limits of long-term strength
					T-12-4.1	Pilot-industrial mastering of materials
			DDN-12-5	Operation properties and characteristics, feasibility of TC mounting and dismantling	T-12-5	Development, fabrication and tests of full-scale TC sample
					T-12-5.1	Creation of test facility and testing of TC prototype
					И-12-5.2	TC testing at test facility for “cold” TM tests
					T-12-5.3	TC testing during start-up and adjustment tests in PCU
12.1 Stator Seal	2.3-113A, Rev.1	Stator Seal TDP	DDN-12.1-1	Data on operability and reliability of TCSSs	T-12.1-1	Creation of seal mockup and verification of its operability in air

System or Component	TDP		DDN		DDN Acquisition Means	
	Product	Title	Number	Need	ID	Summary
					T-12.1-1.1	Development and manufacture of seal prototype and testing under design conditions
					T-12.1-1.2	Development, manufacture of full-scale TC stator seal and acceptance tests
7 Recuperator	02.03-51B	Recuperato r TDP	DDN-7-1	Data on thermohydraulic parameters of upgraded recuperator	T-7-1	Development and manufacture of recuperator heat exchange element model and test facility. Thermohydraulic tests of recuperator heat exchange element model
			DDN-7-2	Data on aerodynamic parameters of recuperator module inlet and outlet sections	T-7-2	Development and manufacture of recuperator module model and test facility. Aerodynamic tests of recuperator module model
			DDN-7-3	Data on operability of recuperator leaky module detection system	T-7-3	Development and manufacture of recuperator leaky module detection system mockup. Recuperator leaky module detection system mockup tests
8 Precooler	2.3-28B	Precooler TDP	DDN-8-1 (exception)		T-8-1 (exception)	
			DDN-8-2	Data on thermohydraulic parameters of cooler cassette	T-8-2	Development and manufacture of test cooler cassette and test facility. Thermohydraulic tests of cooler cassette
			DDN-8-3	Data on vibratory parameters of cooler cassette	T-8-3	Testing of test cooler cassette at vibratory facility
9 Intercooler	2.3-25B	Intercooler TDP	DDN-9-1	Data on aerodynamic parameters of intercooler module inlet and outlet sections	T-9-1	Development and manufacture of cooler module model and test facility. Aerodynamic tests of cooler module model

Table 5-8. DDN Schedule Requirements

DDN Category	Programmatic Need Date	Calendar Date[32]	Logic
C.07.00 FUEL/FISSION PRODUCT			
C.07.01 Fuel Fabrication	Prior to fabrication of qualification test fuel (AGR-5/-6) [Prior to fabrication of initial core][33]	6/13 [6/14]	Qualification fuel must be fabricated with qualified processes. Initial core must be fabricated with qualified processes.
C.07.02 Fuel Performance	Validated fuel performance models 1 yr prior to Operating License[34]	10/16	NRC will not issue Operating License without validated source terms.
C.07.03 Radionuclide Transport	Validated FP transport models 1 yr prior to Operating License[35]	10/16	NRC will not issue Operating License without validated source terms.
C.07.04 Core Corrosion Data	1 year prior to completion of Final Design	10/12	Air and water ingress events are high consequence accidents for MHRs
N.07.05 Spent Fuel Disposal	Prior to Operating License	10/17	Disposition of spent fuel determined prior to generating it in quantity
11.00 REACTOR SYSTEM			
C.11.01 Neutron Control System	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.
C.11.02 Reactor Internals & Hot Duct	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.
C.11.03 Reactor Core	1yr prior to completion of Final Design	10/12	DV&S test results should be factored into Final Design.
C.11.04 Reactor Service Equipment	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.

32 Nominal calendar date consistent with Option 2 NGNP schedule (PPMP 2006); see Section 10.
33 Assuming fuel fabrication facility is built in USA to provide initial NGNP core and reload segments.
34 GT-MHR fuel performance DDNs called for completion of fuel proof test by the end of Final Design (10/13) which is impractical for NGNP.
35 GT MHR FP transport DDNs called for validated models 1 year prior to completion of Final Design (10/12) which is impractical for NGNP.

DDN Category	Programmatic Need Date	Calendar Date[32]	Logic
C.12.00 VESSEL SYSTEM			
C.12.01 Vessels	Beginning of Final Design	10/10	Choice of RPV material has major impact on RS design and long-lead items.
N.13.00 PRIMARY HEAT TRANSPORT SYSTEM			
N.13.01 PHTS Circulator	2 years prior to completion of Final Design	10/11	High temperature circulator is new component for MHR design
N.13.02 IHX	Beginning of Final Design	10/10	IHX is critical component for 950 ^{o}C operation and success of mission.
C.14.00 SHUTDOWN COOLING SYSTEM (SCS)[36]			
C.14.01 Shutdown Circulator	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.
C.14.04 Shutdown Heat Exchanger	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.
C.16.00 REACTOR CAVITY COOLING SYSTEM (RCCS)			
C.16.00	Prior to completion of Final Design	10/13	RCCS is key to maintaining fuel and RPV temperatures during core heatup accidents.
C.21.00 FUEL HANDLING AND STORAGE SYSTEM			
C.21.01 Core Refueling	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.
C.31.00 REACTOR PROTECTION SYSTEM[37]			
C.31.01 Safety Protection and Instrumentation	Prior to completion of Final Design	10/13	Instrumentation and controls can be modified prior to startup if necessary
C.34.00 PLANT CONTROL, DATA AND INSTRUMENTATION SYSTEM			

[36] GT-MHR DDNs call for completion at the start of Final Design (10/10)
[37] GT-MHR DDN calls for completion at the beginning of Final Design (10/10)

DDN Category	Programmatic Need Date	Calendar Date[32]	Logic
C.34.01 Nuclear Island Control and Instrumentation	Beginning of Final Design	10/10	Plateout probe needs to be incorporated into plant design.
N.41.00 POWER CONVERSION SYSTEM (Table 5-7)			
Turbomachine	Prior to completion of Final Design	10/13	Turbine is high-risk FOAK component for 950 °C operation and for success of mission.
Recuperator	Prior to completion of Final Design	10/13	Recuperator is high-risk FOAK component.
Precooler/Intercooler	Prior to completion of Final Design	10/13	DV&S test results should be factored into Final Design.
N.42.00 SECONDARY HEAT TRANSPORT SYSTEM			
N.42.01 SHTS Circulator	1 year prior to completion of Final Design	10/12	High temperature circulator is new component for MHR design
N.42.02 Isolation Valves	1 year prior to completion of Final Design	10/12	High temperature isolation valve is new component for MHR design
N.44.00 SI-BASED H2 PRODUCTION SYSTEM			
N.44.01 Sulfuric Acid Decomposition	1 year prior to completion of Final Design of NGNP SI plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.
N.44.02 Bunsen Reaction	1 year prior to completion of Final Design of NGNP SI plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.
N.44.03 HI Decomposition	1 year prior to completion of Final Design of NGNP SI plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.
N.45.00 HTE-BASED H2 PRODUCTION SYSTEM			
N.45.01 SOE Cells	1 year prior to completion of Final Design of NGNP HTE plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.
N.45.02 SOE Units	1 year prior to completion of Final Design of NGNP HTE plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.
N.45.03 SOE Modules	1 year prior to completion of Final Design of NGNP HTE plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.

DDN Category	Programmatic Need Date	Calendar Date[32]	Logic
N.45.04 HTE Plant Supporting Equipment	1 year prior to completion of Final Design of NGNP HTE plant	10/12	FOAK plant. Results from Eng-Scale Pilot Plant needed for Final Design.

6. REQUIRED TECHNOLOGY DEVELOPMENT PROGRAMS

The DOE-sponsored technology programs intended to support the NGNP, including the various NGNP R&D programs and the NHI programs, have been evaluated, and their responsiveness to the NGNP DDNs (Section 5) was assessed. The existing TDPs were critiqued on an exception basis to identify any deficiencies and unnecessary workscope, especially in the context of the NGNP schedule. In general, these TDPs propose to investigate an excessively large number of materials (graphite, metals, etc.) and so the candidate materials need to be prioritized.

The results of these assessments are summarized in Table 6-1[38] and discussed in the following subsections. In large measure, this section is a critique of the existing TDPs - consistent with the purpose of providing focus and prioritization to the NGNP R&D programs - rather than a stand-alone, bottoms-up TDP. While this approach is considered appropriate and sufficient for preconceptual design, an integrated, umbrella NGNP TDP should be prepared during the Conceptual Design phase and periodically updated as the NGNP design matures and feedback is obtained from the licensing authorities and potential customers.

6.1 Fuel/Fission Products Program

The DOE/NE AGR Fuel Development and Qualification Program (AGR Plan/1” 2005) has the mission to develop and qualify fuel for the NGNP (PPMP 2006).[39] Its stated goals are:

- Provide a baseline fuel qualification data set in support of the licensing and operation of a VHTR (e.g., NGNP demonstration plant). Gas-reactor fuel performance demonstration and qualification comprise the longest duration R&D task for VHTR feasibility. The baseline fuel form is to be demonstrated and qualified for a peak time-averaged fuel centerline temperature of 1250 ^{o}C
- Support near-term deployment of a VHTR for hydrogen and energy production in the United States (2020) by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Using international collaboration mechanisms, extend the value of DOE resources.

The AGR Fuel Program is developing and qualifying conventional, SiC-based TRISO fuel particles with the assumption that conventional TRISO particles will be adequate for use in the initial core of the NGNP. However, there was no NGNP reference design when the AGR Fuel Program was first planned in 2003, and so the program is effectively a generic program. The program chose to qualify an LEU, TRISO-coated, 350-μm UCO particle with the German TRISO-coating system – effectively the commercial GT-MHR fissile particle (Munoz 1994) with

[38] The tables are located at the end of the section.
[39] Other fuel development program plans have been prepared under the Advanced Fuel Cycle Initiative (AFCI) which could ultimately be relevant to the NGNP: (1) a plan for the development of UO_2^* (Zr-buffered, TRISO-coated UO_2 particles) and ZrC-coated particles (Hanson 2004a), and (2) a plan for the “deep burn” of recycled LWR spent fuel in MHRs (DB-MHR Plan 2002).

an improved TRISO coating design. Validation of radionuclide source terms is also within the scope of the AGR Fuel Program.

The AGR fuel plan has the following major program elements:

- Fuel manufacture - This element addresses the work necessary to produce coated-particle fuel that meets fuel performance requirements and includes process development for kernels, coatings, and compacting; quality control (QC) methods development; scale-up analyses; and process documentation needed for technology transfer. This effort will produce fuel and material samples for characterization, irradiation, and accident testing as necessary to meet the overall goals. The plan also identifies work to develop automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost; that work will be conducted during the later stages of the program in conjunction with cosponsoring industrial partners.

- Fuel and materials irradiation - The fuel and materials irradiation activities will provide data on fuel performance under irradiation as necessary to support fuel process development, to qualify fuel for normal operation conditions, and to support development and validation of fuel performance and fission product transport models and codes. It will also provide irradiated fuel and materials as necessary for PIE and safety testing. A total of eight irradiation capsules have been defined to provide the necessary data and sample materials.

- Safety testing and PIE - This program element will provide the facilities and processes to measure the performance of AGR fuel systems under normal operating conditions and accident conditions. This work will support the fuel manufacture effort by providing feedback on the performance of kernels, coatings, and compacts. Data from PIE and accident testing will supplement the in-reactor measurements [primarily fission gas release-to-birth ratio (R/B)] as necessary to demonstrate compliance with fuel performance requirements and support the development and validation of computer codes.

- Fuel performance modeling - Fuel performance modeling, as defined in the context of this plan, addresses the structural, thermal, and chemical processes that can lead to coated-particle failures. It does not address the release of fission products from the fuel particle, although it considers the effect of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Computer codes and models will be further developed and validated as necessary to support fuel fabrication process development and plant design and licensing.

- Fission product transport and source term - This element will address the transport of fission products produced within the coated particles to provide a technical basis for source terms for AGRs under normal and accident conditions. The technical basis will be codified in design methods (computer models) validated by experimental data, as necessary to support plant design and licensing.

The AGR program includes eight irradiation tests in the ATR to address the DDNs related to fuel performance and fission product release under irradiation and to provide irradiated fuel specimens for postirradiation heating (accident simulation) tests. These tests are summarized in Table 6-2 (AGR Plan/0 2003).

The most recently published schedule (as of FY06) for the AGR program major activities is shown in Figure 6-1. The estimated total cost is 193 $M (FY06 dollars); the planned spending profile is illustrated in Figure 6-2.

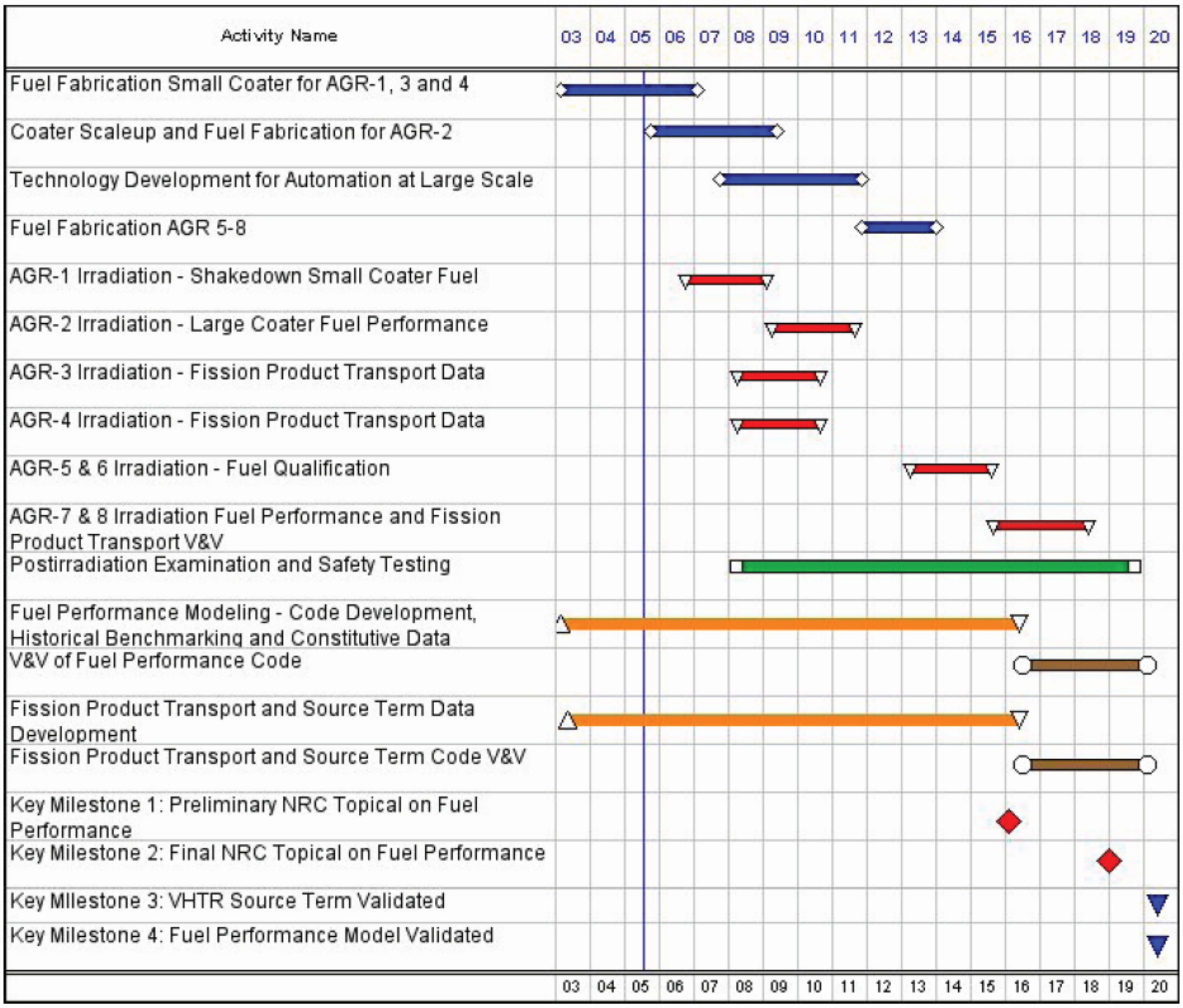

Figure 6-1. AGR Program Schedule (2005)

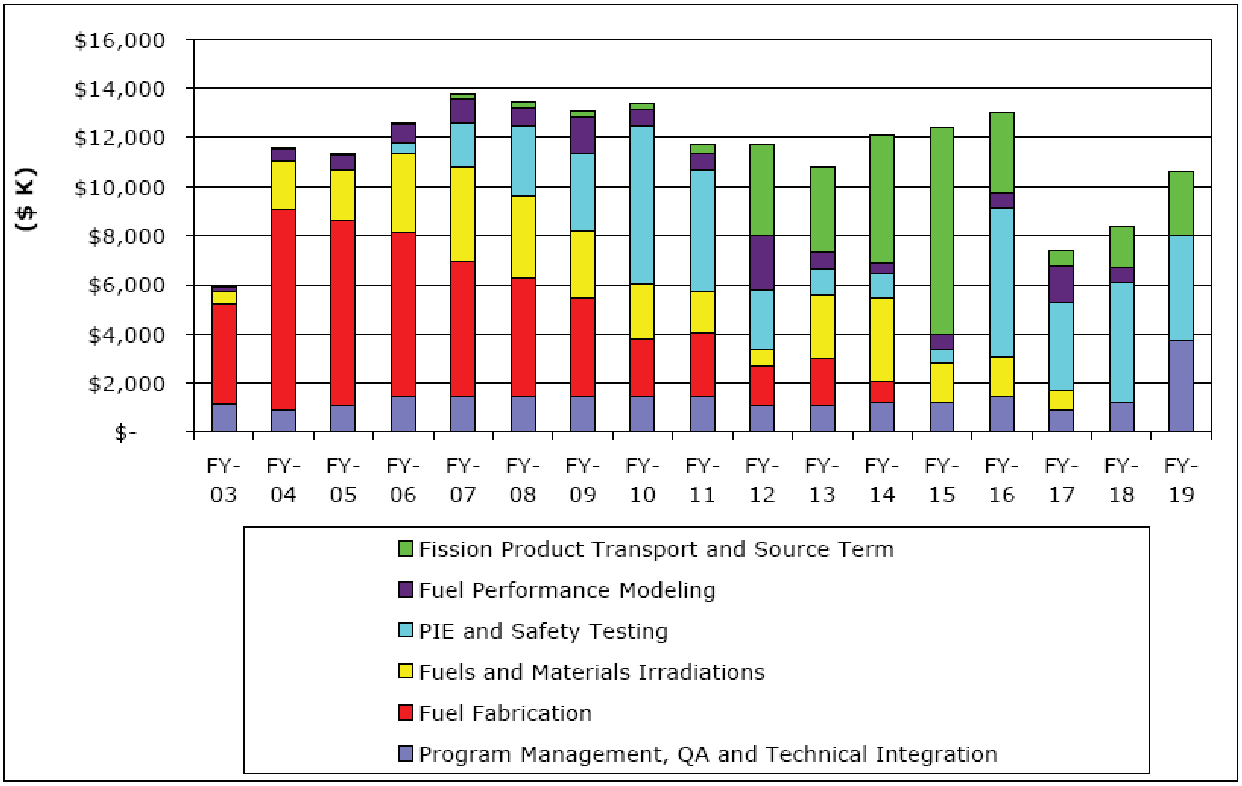

Figure 6-2. AGR Program Planned Expenditures

The AGR program plan (AGR Plan/1 2005) is a comprehensive plan[40] which the GA team continues to endorse with the caveats summarized in the following subsections.[41] The PPMP (2006) provides a good summary of the AGR Plan. The PPMP also identified a number of risks associated with the overall NGNP fuel qualification effort which can be summarized as follows:

- TRISO-coated particle fuel cannot be qualified to meet the aggressive AGR performance envelope and support the NGNP schedule logic.
- The safety case has not been demonstrated at burnups beyond ~10% FIMA for either UO_2 or UCO TRISO-coated fuel.
- The dual fabrication path selected may not minimize schedule risk associated with fuel qualification. The use of the foreign fuel vendor (NFI) has a number of attendant technical risks.
- There is limited irradiation capacity and capability in ATR to meet NGNP (flux level, spectrum, and physical size) fuel needs. Similarly, the amount of PIE and safety testing to be performed may represent a schedule risk if performed at one facility.
- Hot-cell facilities at INL do not adequately support post irradiation examination of fuel being developed in support of NGNP.

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[40] The initial issue ("AGR Plan/0" 2003) contained several appendices that provided useful background information and additional programmatic detail regarding scope, schedule and cost.
[41] GA was actively involved in the original development of the AGR program plan and continues to participate in the program.

- The use of a foreign (non-NRC licensed) fuel vendor presents a regulatory risk.
- R&D necessary to qualify the source term for NGNP may not meet requirements for the licensing basis.
- The lack of an NGNP fuel design is impacting fuel qualification schedule for NGNP.
- In order to accelerate the schedule, large coaters will need to be used as soon as possible in the program for fuel fabrication. However, only small coaters are currently available.

The ITRG (2004) expressed similar concerns with regard to fuel development and qualification. In general, the GA team agrees that these are legitimate programmatic concerns and is supportive the proposed actions in the PPMP to minimize them. GA has identified further issues with the AGR program which are described below.

6.1.1 Fuel Process Development

GA proposes to use 10%-enriched UO_2 fuel fabricated by NFI for the NGNP first core and possibly for one or more reload segments. Accordingly, new DDNs have been included in Table 5-6 that are consistent with this approach. However, GA views this as a necessary (and somewhat undesirable) option only to allow startup of the NGNP by 2018 because the NGNP Project must develop a domestic supply of high-burnup UCO coated-particle fuel (assuming that the NGNP is a prismatic block MHR) in order to meet the NGNP project objectives as identified in the NGNP PPMP. It is also questionable whether the “proof-of-principal” operating period starting in 2018, which is to serve the purpose of providing “…the basis for commercialization decisions by industry…” can be accomplished using Japanese UO_2 fuel. Thus, procuring UO_2 fuel from NFI indefinitely will not be acceptable, and development of UCO fuel should continue as a high priority under the AGR Fuel Program.

The project objectives stated in the NGNP PPMP appear to acknowledge the necessity of developing a domestic fuel supply, but there is no activity included in the NGNP project plan to do so nor is such scope included in the AGR Fuel Program (now part of the NGNP Project). As written, the AGR Plan/1 (2005) is designed to qualify fuel, not to develop the technologies to economically mass produce it. Presumably, the NGNP Project therefore expects that private industry will build a commercial fuel manufacturing plant capable of supplying the fuel for the NGNP. However, GA considers it unlikely that US industry would be willing to invest many millions of dollars to establish a fuel manufacturing facility with the necessary capacity to make the NGNP initial core within a two to three year time frame, given the uncertainty with respect to the potential for, and time frame of, any follow-on coated-particle fuel business.

Consequently, GA believes that it is essential that the NGNP Project build, license, and operate a fuel manufacturing pilot plant for NGNP to demonstrate the viability of economical mass production of coated-particle fuel, thereby satisfying the fuel fabrication process DDNs identified in Table 5-1. More specifically, GA recommends that an NGNP Fuel Fabrication Facility (FFF) be built in Idaho to supply the fuel for the NGNP. The NGNP FFF should be designed for a full production capacity of 510 fuel elements per year. The facility would be operated at full capacity for two years to produce the initial core and the production rate would then be reduced to 340 fuel elements per year, at which rate the facility would produce a reload segment every eighteen months. To support the NGNP Project Option 2 schedule and the GA NGNP fuel acquisition strategy, the NGNP FFF would have to be designed, built, licensed, and qualified from 2013 through 2019 and used to fabricate the second core fuel load (to replace the NFI fuel used for the first core fuel load) in 2020 and 2021.

The 510 fuel element/year process line that would be built and demonstrated in the NGNP FFF during production of the second NGNP core fuel load would be the basic production module that would be replicated in a commercial fuel fabrication facility. Thus, the NGNP would demonstrate the fuel fabrication technology needed for the commercial fuel supply business, thereby greatly reducing the costs and risk that would be associated with a first-of-a-kind facility. This assertion is, of course, based on the premise that the US government would make the NGNP pilot line technology available to any US company that wishes to replicate the technology to develop a commercial MHR fuel manufacturing business.

Another issue with respect to fuel process development is coater scale-up. The fuel currently being irradiated in AGR-1 was made in a laboratory-scale coater at ORNL. Coating process development is currently proceeding at BWXT to scale up the coating process to a 15-cm diameter coater. Commercial scale coaters operated at GA and at HOBEG GmbH in Germany had a diameter of 24-cm. The AGR Fuel Program understands that a 15-cm diameter coater is not of sufficient size for commercial fuel production but has elected to scale-up the coating process in two steps. Under the AGR program approach, the fuel for the AGR-2 fuel demonstration test will be made in the 15-cm diameter coater, but a second scale-up will be necessary, presumably prior to making the qualification test fuel for irradiation tests AGR-5 and AGR-6. However, this second coater scale-up activity is not defined in the current AGR Plan/1, and the schedule and funding profiles in the Plan do not account for it.

6.1.2 Fuel Materials Qualification

Both the ITRG (2004) and INL (PPMP, 2006) have recognized the risks associated with the AGR Fuel Program's single path approach to fuel qualification. Indeed, (PPMP 2006) calls for expansion of the program to include a dual path involving irradiation testing of UCO fuel fabricated in the USA by BWXT and UO_2 fabricated by NFI. The UCO fuel would be irradiated in AGR-2 as originally planned, and a new irradiation test, "AGR-2a," would be added to the program plan for irradiation testing of UO_2 fuel fabricated by NFI. Irradiated fuel from both irradiation tests would be subjected to heating tests to simulate accident conditions (i.e., safety tests). The cost of adding the NFI UO_2 path to the program was estimated to be about $17M. In the approach described in the PPMP, a down selection would be made based on the irradiation performance and safety test results from AGR-2 and AGR-2a, and only one of the two candidate fuels would be subjected to qualification testing in irradiation tests AGR-5 and AGR-6[42].

GA endorses the approach described in the NGNP PPMP to irradiate UCO fuel and NFI UO_2 fuel in AGR-2 and AGR-2a, respectively. However, consistent with GA's view that demonstration of UCO fuel in the NGNP is essential for deployment of commercial MHRs in the USA, GA does not agree that a down selection for qualification testing in AGR-5 and AGR-6 should be made between UCO fuel and NFI UO_2 fuel. Rather, UCO fuel should be qualified in AGR-5 and AGR-6 as currently planned, and NFI UO_2 fuel should be qualified for use in NGNP based on Japanese irradiation and safety test data, proof testing in AGR-2a, and fuel performance monitoring, as necessary, in the NGNP.

6.1.3 Radionuclide Transport

As indicated in the PPMP (2006), there is a substantial risk that the RN transport workscope included in the AGR Plan/1 (2005) will be inadequate to support NGNP design and licensing. This problem has been exacerbated by chronic funding shortfalls for the AGR program; consequently, no experimental work in the RN transport area has been initiated with the exception that the driver fuel has been fabricated for irradiation tests AGR-3 and AGR-4. In fact, no experimental work on RN transport outside of the core is planned until FY12. The significant RN transport issues identified with the AGR Plan are summarized below.

6.1.3.1 RN Release from the Core

Two types of tests are needed to characterize RN release from the reactor core: (1) single-effects tests to generate differential data for deriving improved component models and material property correlations, and (2) independent integral tests to confirm the validity of the upgraded design methods. In the AGR Plan, the single-effects tests are identified as AGR-3 and AGR-4, and the validation test is identified as AGR-8. In essence, the overall objective of these tests is to characterize FP transport in fuel kernels, particle coatings, fuel-compact matrix and fuel-element graphite. However, the typical irradiation capsule design is ill-suited for this purpose, and this limitation applies to the AGR-1 capsule design developed for fuel irradiation testing in the AGR Fuel Program.

In order to satisfy the DDNs related to RN release from the core and to address the limitations of past experimental efforts, an experimental program has been recommended that is comprised of three types of tests: (1) irradiation tests with a known fission product source, (2) postirradiation heating tests, and (3) laboratory-scale sorption measurements for fuel-compact matrix and graphite (Hanson 2005b). A different capsule geometry for AGR-3, AGR-4 and AGR-8 was also proposed that should facilitate the determination of FP transport properties

[42] The Technical Program Plan for the AGR Fuel Program (AGR, 2005) has not been updated to reflect the dual path approach described in the NGNP PPMP. Indeed, an INL-led team recently conducted a survey of potential fuel vendors in order to develop a fuel acquisition strategy for the NGNP. The results of the study have not been released to the public at this writing.

(e.g., diffusion coefficients, etc.). Sorption measurements are not explicitly described in the AGR Plan, but it does state that the sorptivities of fission metals on fuel-compact matrix and fuel-element graphite should be determined. Since there is no practical way of deriving sorption isotherms from irradiation capsule data, separate laboratory-scale measurements are necessary; a test specification has been prepared (Hanson 2006b). AGR program management has been receptive to these recommendations, and it is anticipated that they will be incorporated into the next revision of the AGR Plan.

6.1.3.2 RN Transport in the Primary Circuit

The AGR Plan identifies the need to perform a series of single-effects RN transport tests in order to generate the differential test data to provide the bases for deriving improved component models and material property data correlations for predicting RN transport in the primary coolant circuit. Once these out-of-pile, single-effects tests have been performed, independent integral tests in an in-pile loop need to be performed to confirm the validity of these upgraded design methods. However, the AGR Plan provides minimal definition of these test programs, and no experimental work is planned until FY12.

A single-effects experimental program has been recommended (Hanson 2005a) that is comprised of three types of tests: (1) laboratory-scale sorption measurements, (1) out-of-pile loop tests, and (3) decontamination tests. The deposition behavior of four radionuclides (Ag, Cs, Te, and I) with different chemistries on at least three different metals will be determined as a function of temperature, partial pressure, and coolant chemistry (which determines surface oxidation state). The number of sorption measurements (equilibrium surface concentration as a function of temperature and partial pressure) needed to derive reliable isotherms is large. A test specification for the sorption measurements has been prepared (Hanson 2006b).

Relying exclusively upon loop tests to produce all of the required data would be too expensive and time consuming; consequently, much of these data will be generated in simpler lab-scale test facilities. Nevertheless, out-of-pile loop tests are also necessary to investigate the effects of flow and high system pressure and to perform *in situ* liftoff tests, and a series of such loop tests have been proposed (Hanson 2005a). Finally, decontamination tests will be performed on contaminated specimens produced from both the sorption tests and the loop tests.

A series of fission product transport tests in an in-pile loop are needed in order to generate the integral test data necessary to validate the predicted source terms for the NGNP. To that end, the functional and technical requirements for an in-pile fission product transport loop to support VHTR design and licensing have been identified and systematically documented (Hanson 2004b). The AGR Plan/1 (2005) contains tasks to construct an in-pile loop and to perform an in-pile test program. However, the design and construction of the loop are not initiated until FY13. The technical feasibility of constructing such a facility (presumably in the ATR) and the attendant costs and schedule must be established far earlier if the design methods for predicting

RN in the primary circuit are to be validated on the required schedule. In addition, the cost and schedule estimates for loop design and construction appear to be extremely optimistic.

6.1.3.3 RN Transport in VLPC

As described previously (Section 3.2.2.3), credit is taken for RN retention in the VLPC during core heatup accidents. No direct measurements under MHR VLPC conditions have been made. The AGR/1 Plan calls for an evaluation of the extent to which the extensive international database for RN transport in water-reactor containments might be applied to refine and to independently validate the design methods used for predicting radionuclide transport in the VLPCs of modern MHRs.

Such an evaluation was recently made (Hanson 2007). It concluded that the experimental water-reactor database for radionuclide transport in containment buildings is of limited value for refining and independently validating the design methods used to predict radionuclide transport in VLPCs because the radionuclide concentrations and physical and chemical forms in the two systems are too different. Consequently, new DDNs have been identified (Table 5-6) which the AGR program needs to address.

6.1.3.4 Tritium Transport Behavior

As described previously (Section 3.2.2.4), tritium will be generated in the NGNP by various nuclear reactions, and some of this tritium will accumulate in the primary helium coolant. Moreover, a fraction of the tritium in the primary coolant will permeate through heat exchanger walls and contaminate the product hydrogen. This tritium contamination represents a potential radiation hazard to the plant workers and to the consumers; consequently, the level of tritium contamination must be characterized and controlled to acceptable levels.

The AGR/1 Plan does not address tritium transport (perhaps, in part, because it is a generic development plan which does not focus on a specific reactor design). Tasks to characterize tritium retention in the core and tritium permeation through heat exchanger materials (Hanson 2006a) need to be added to address NGNP DDNs.

6.2 Materials Programs

The objective of the NGNP Materials R&D Program (2005) is to provide the essential materials R&D needed to support the design and licensing of the reactor and balance of plant, excluding the hydrogen plant. The most important products of the program will be qualified nuclear graphite for the reactor core and high temperature metals for use throughout the nuclear heat source, power conversion system, primary heat transport system, and balance of plant. The GA team perspective on the graphite and metals program is summarized below.

The preparation of a plan for the definition, development, and characterization of graphite, ceramic, and metallic materials for NGNP components is an impressive effort, requiring the skills of many materials experts across the country. Notably, this effort has been performed at a time when candidate designs for the NGNP have been in a preconceptual design phase. Nevertheless, this R&D plan appears to be generally responsive to the NGNP materials DDNs identified in Section 5 with the important exceptions noted below.

6.2.1 Core Graphites

The graphite program described in the NGNP Materials R&D Program Plan is evaluating at least 16 nuclear graphites and fuel-element matrix materials from at least four international graphite vendors. The current focus of the program is the graphite irradiation capsule AGC-1 which is intended to provide irradiation creep- and dimensional change data on candidate graphites for the use in the NGNP. Creep data will be obtained for six major graphite grades (vendor in parenthesis): H-451 (SGL) and IG-110 (Toyo Tanso), both of which are included as reference graphites, and four new grades, PCEA (Graftech), NBG-17 (SGL), NBG-18 (SGL), and IG-430 (Toyo Tanso). In addition, AGC-1 contains 10 minor grades of graphite.

The complete characterization of all selected candidate graphites is believed to be cost prohibitive. Therefore, it is recommended that the number of selected grades be reduced to a more realistic number. Some attention should be paid to first selecting those grades which were previously qualified for use in a previously operated HTGRs (e.g., grades PGX and HLM for FSV) or currently being used in operating HTGRs (e.g., IG-110 in HTTR and HTR-10). These grades have substantial databases which simply need to be extended to cover the higher temperatures in the NGNP compared to previous HTGRs.

A comprehensive, stand-alone graphite TDP is urgently needed which defines the entire scope, schedule and cost of the planned program. The planned program is probably responsive to the graphite DDNs defined herein for a prismatic NGNP, but it may be excessive from the GA team perspective. The graphite service conditions in a prismatic VHTR are not demanding (e.g., fast neutron fluence to the fuel-element graphite is $<5 \times 10^{21}$ n/cm^2, E >0.18 Mev). Previously qualified H-451 for fuel and reflector elements and Stackpole 2020 for the core support structure have adequate material properties.

From the GA team perspective, the primary requirement for the NGNP Project is to identify and qualify a replacement graphite for H-451. The recommended approach is to use AGC-1 as a screening capsule to identify the lowest-cost graphites with properties comparable to H-451 and then to perform supplemental testing to establish a correspondence between the behavior of the replacement graphite and the extensive H-451 experience base. While it is important to minimize the number of graphites to be characterized, two or more domestic suppliers of H-451 replacement graphite should be qualified. The GA team considers the qualification of a replacement graphite for H-451 to be a high priority, but a low risk, task.

These irradiations are being planned only in the USA although information regarding some of the grades will be forthcoming from irradiations sponsored in other countries (e.g., HTR-TN irradiations in HFR Petten). With the demanding requirements for irradiation tests in this program, other foreign irradiation test facilities should be considered (e.g., Russia and Japan).

6.2.2 C-C and SiC-SiC Composite Materials

The program to address the qualification and testing of C-C and SiC-SiC composite materials appears to be excessive, especially for SiC-SiC composite materials. The development of C-C composite materials has matured extensively over the past years, but SiC-SiC composite materials development is still an immature technology. These materials have been selected for primarily for replacing metallic materials in control rod components which will see high temperatures and high fluences that will probably not allow them to remain in the reactor environment for full lifetime. However, they have been designed to be replaced at acceptable intervals. Efforts to select more easily fabricable metallic candidate materials and characterize their high temperature and environmental responses should be emphasized. Composite materials efforts, especially those regarding the use of C-C composites should be continued, but with only minor efforts with regard to SiC-SiC composites.

6.2.3 High-Temperature Metals

The following comments generally follow the organization of the NGNP Materials R&D Plan.

6.2.3.1 Selection of High-Temperature Metallic Materials

The high-temperature metallic materials efforts appear to be responsive overall to the NGNP DDNs, capturing the specific properties required for design with several exceptions. The selection of candidate materials for the NGNP, which is still in its preconceptual design phase, has primarily centered on wrought alloys with Alloy 617 as a primary candidate, a good choice based on a fairly extensive database for this alloy under near reactor conditions. Other candidate alloys selected include other wrought alloys, which are reasonable alternates for high-temperature components (Alloy 800H, Hastelloy X, Hastelloy XR, Haynes 230, Haynes 214, etc.), some of these alloys also have extensive databases developed under near prototypical conditions. However, candidate materials are not well defined for some components within the PCS, especially for the turbine. No materials have been defined for turbine blades, and no program has been planned to specify candidates and characterize same. Excellent candidates for the turbine blades include IN 100 and IN 738; these choices are based upon a reasonably extensive database obtained under HTGR conditions.

6.2.3.2 Expansion of ASME Codes and ASTM Standards

This effort has been well planned and is in concert with the efforts for development, characterization, and qualification of graphites, ceramics, and metallic materials in the R&D programs.

6.2.3.3 Thermal Aging and He Effects on Metals

This area is generally responsive to the metallic materials DDNs and is a well-planned comprehensive effort. However, considering that previous research has shown that materials can degrade via interaction with helium impurities at elevated temperatures (e.g., carburization, decarburization, etc.), some attention should be given to determining the reactor environmental regime (temperature, impurities levels, etc.) which might be benign for those materials used at the highest temperature where such effects may be maximized. Parametric studies to determine the effects of varying impurities of extreme importance (e.g., H_2O and CH_4) should be included and consideration given to pre-exposure of materials prior to environmental testing to develop protective scales to minimize these interactions with attendant tests to determine the stability of those protection techniques. Also, some consideration should be given to the possibility of doping the primary coolant (as is done in the British CO_2-cooled AGRs).

6.2.4 Reactor Pressure Vessel Materials

The recommended RPV material for NGNP is 2¼Cr-1Mo with 9Cr-1Mo-V as a backup (and potential product improvement for commercial H2-MHR).

6.2.4.1 Irradiation Testing and Qualification of Candidate RPV Materials

With the exception of discussions regarding the development and construction of a new irradiation test facility to replace the Ford Test Reactor at the University of Michigan, no planning appears to be in place for providing irradiation data for qualifying RPV materials. Such a plan is required. Also, there is no indication that consideration has been given to the possible use of foreign test facilities (EU, Russia, Japan, etc.) for obtaining RPV irradiation data in a sufficiently timely manner for meeting the NGNP schedule.

6.2.4.2 Emissivity and other Physical and Mechanical Properties of Vessel Surface

The need for these data has been defined and methods for achieving the solution to this issue indicated, but no specific program plan for fulfilling the need for these data was presented.

6.2.5 Development of a Materials Handbook/Database

This effort appears to be well intentioned, but it needs to specifically include data from published reports from previous gas reactor materials programs which were designed and performed to specifically address the issues of primary coolant compatibility and thermal aging effects. It is uncertain that all information in this regard will be captured. This impression is based upon discussion in the NGNP R&D Plan (2005) regarding past experimental efforts in these regards, where efforts in foreign countries, at US National Laboratories, and at some, but not all, private companies (notably GE) were described. Most notably, there was no mention of the large gas-cooled reactor materials screening programs at GA, in conjunction with HTMP in the UK, EIR in Switzerland, and KFA and supporting companies (BBC, HRB) in Germany. There is a compelling need to include all of these past data.

6.2.6 Materials Issues Associated with the NGNP Power Conversion Unit

No definite plans for a materials test program in support of PCS component materials is in place. Candidate materials do not include any materials for turbine blades, a significant deficiency. Also, the Plan doe not indicate any significant issues with regard to the recuperator materials. The recuperator may be a fin-plate design with very thin sections. Even though the thin materials may operate at what may be considered low temperatures (<600 ^{o}C), the thin materials which may be used in its construction should be examined with respect to degradation by primary coolant impurities.

6.2.7 RPV Fabrication and Transportation Issues

Candidate materials have been identified for the RPV (as well as other vessels in the system), and concerns regarding their fabrication and transportation to the construction site have been identified but not in any detail. Specific plans for materials R&D are apparently not yet in place. These details, such as the characterization of candidate RPV materials for irradiation response, weldability, post-weld heat treatment requirements, etc., need to be more specifically addressed.

6.3 Intermediate Heat Exchanger Program

The materials development for the IHX is included in the NGNP Materials R&D Plan (2005). The GA team understands that a separate Energy Transfer TDP will be prepared that will presumably address the non-materials development needs for the IHX (e.g., confirm temperature distribution, etc.).

As an indication of the expected content of the Energy Transfer TDP, the Toshiba IHX TDP is summarized in Figure 6-3. The approach of the Toshiba TDP is to produce a PCHE core applicable to the NGNP IHX using approved materials, test the strength of the diffusion bonded joints, and perform heat transfer and thermal cycling tests which are the basis for analytical evaluation and non-destructive and destructive examinations to confirm the validity of the design. The program is projected to take approximately four years to complete. The future availability of these Toshiba data to the NGNP Project is unclear at this writing. Anticipated intellectual property issues would have to resolved.

Activity ID	Activity Description	Oring Dur	Rem Dur	%	Early Start	Early Finish
WBS C.Q.10.2.32.HTD	Conceptual design					
					27-Feb-07	31-Oct-08
WBS C.Q.10.2.33.HTD	Preliminary design					
					31-Oct-08	31-Oct-10
WBS C.Q.10.2.34.HTD	Final design					
					31-Oct-10	31-Oct-13
R&D	(1) The confirmation of the PCHE core strength				1-Sep-07	31-Oct-10
	(2) The validity confirmation of the design evaluation procedure				1-Sep-08	31-Oct-11
	(3) The confirmation of the PCHE core temperature distribution				1-Sep-08	31-Oct-11
	(4) The validity confirmation of the thermal hydraulic characteristics as a whole IHX				1-Sep-08	31-Oct-11

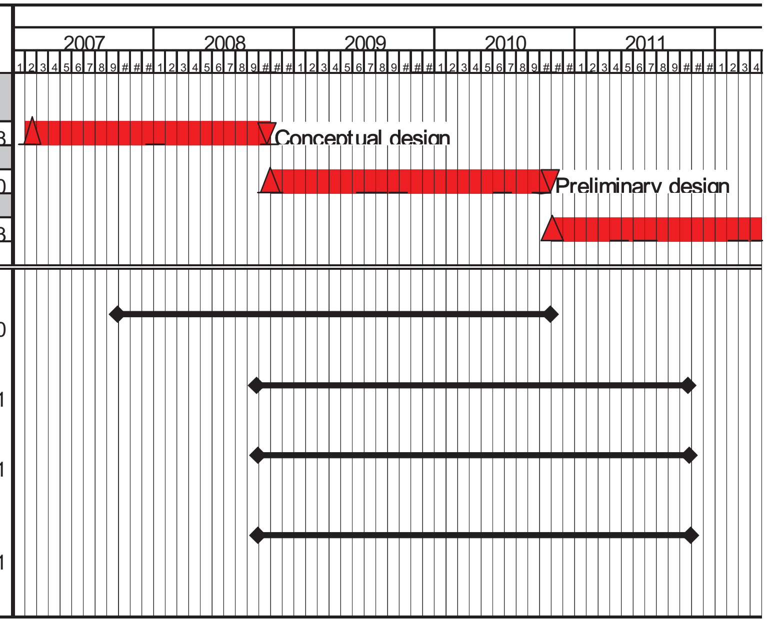

Figure 6-3. Toshiba TDP for Intermediate Heat Exchanger

6.4 Power Conversion System Program

The RF PCU technology demonstration program was jointly developed by OKBM and US program participants (PCU TDP 2005). The GA team believes that this program is capable of qualifying the OKBM PCU design. The design will need to be modified and the TDP supplemented for 950 ^{o}C operation.

The overall logic for the PCU TDP is illustrated in Figure 6-4.

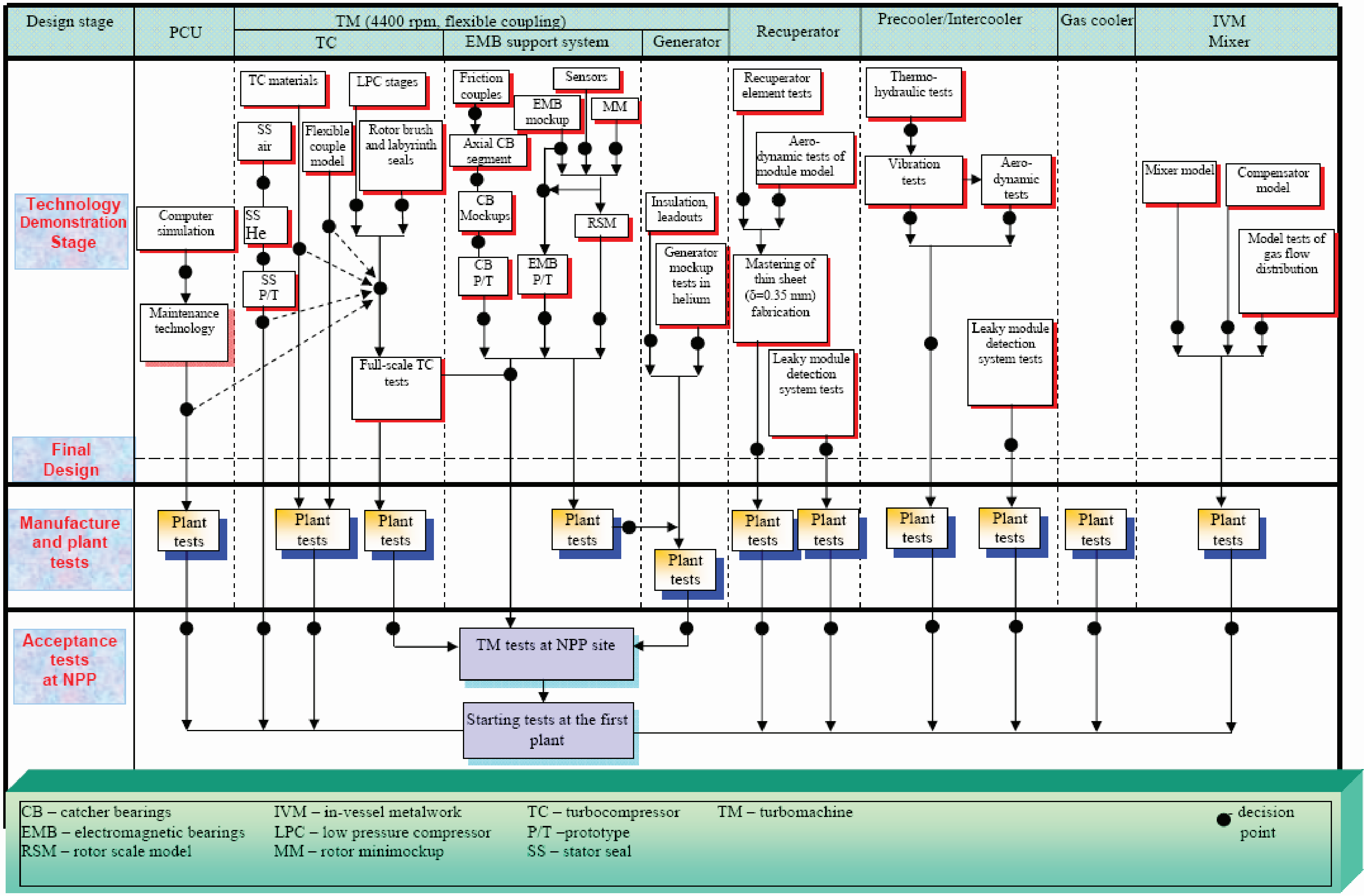

Figure 6-4. Overall Logic for OKBM PCU Technology Development Program

6.4.1 Technology for 850 °C Operation

The purpose of the PCU TDP is to mitigate risk by addressing the following technical objectives:

1. Demonstrate performance characteristics of the **Generator**, including its electrical insulation in helium, and its electrical lead-outs through the PCU pressure vessel. The scope for this development work includes:
    a. Manufacturing a scaled mock-up of the generator. This mock-up will be used to verify the computer models for calculating the temperatures of the generator inner structure cooled by helium, to characterize the use of electromagnetic bearings within a generator including the influence of the generator magnetic field on the magnetic bearing control system, and to check design requirements related to the generator vertical configuration.
    b. Tests of representative insulation samples will be carried out in a multipurpose test facility that will also be used for tests of electrical leadouts. The design of the Insulation and Lead-out Test Facility will allow simulation of the full range of temperatures and pressures seen during operation, plus pneumatic test pressures, up to 11.6 MPa. It will also allow the simulation of rapid depressurization transient.
    c. The generator lead-outs must transfer 344MVA/292MW of power at 20kV, with currents of up to 10kA through penetrations in the PCU pressure vessel. The normal operating pressure inside the vessel is 2.63MPa; however, to enable pneumatic pressure testing of the PCU vessel, the lead-outs must be capable of test pressures up to 11.6 MPa. While electric lead-outs, based on soldered ceramic insulators, are well established, there is no experience at the voltage and power levels required for the GT-MHR
2. Demonstrate the performance characteristics of the **Turbocompressor** high temperature structures at the turbine inlet, of the compressor stages and rotating seals and of the static seals in the turbocompressor stator. The scope for this development work includes:
    a. The selection and qualification of materials for the high temperature structures of the turbine have been initiated through laboratory studies under ISTC Project 1313.[43] ISTC 1313 includes the evaluation and selection of candidate materials and long-term static and cyclic strength tests of structural materials. Additional near-term work is planned to develop an extrapolation procedure for alloy material strength characteristics at working temperatures for the specified 60,000-hr lifetime, to evaluate the allowable cobalt content in metal alloys for turbocompressor structural elements, and to evaluate the potential use of thermal barrier coatings on the turbine blades. The selected materials will be used to develop manufacturing techniques for semi-finished components that are representative of those to be used in the TC.

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[43] The International Science and Technology Center (ISTC) was established by international agreement in 1992 as a nonproliferation program (http://www.istc.ru/). The ISTC coordinates the efforts of numerous governments, international organizations, and private sector industries, to provide weapons scientists from Russia and the Commonwealth of Independent States new opportunities in international partnerships.

b. A scale-model representative of compressor stages will be fabricated and tested in a dedicated facility. The compressor model will correspond to the first three stages of the low-pressure compressor. Its features will include the three stages, along with the shaft and stator casing, plus the inlet and outlet channels. The compressor model will be driven by an electric motor with a nominal speed of 3000 rpm. The output of the motor will be increased from 3200 rpm to 26,000 rpm through the use of a multiplier. Testing will be done first in air and then in helium.

c. The rotating seals to be modelled include the seal that isolates the clean helium in the generator enclosure from the primary coolant and the seals located at the inlet and outlets of the compressor and turbine stages. The performance of these seals will be measured in helium at temperatures and pressure representative of GT-MHR operation.

d. The turbocompressor stator seals tests will verify their performance under reactor operating conditions. The scope of these tests will include first testing in air of a 1000-mm (inside diameter) mock-up of the seals, followed by the testing of an improved version of the seal design (using the available test results) in air. The smallest stator seal prototype will be built using the results from the previous tests and its performance will be verified in helium under reactor operating conditions. After successfully completing the testing of the smallest seal prototype, the remaining five larger stator seals will be built and acceptance tests performed under similar conditions.

3. Demonstrate the performance characteristics of the **Electromagnetic Bearings (EMB) Support Subsystem** with regard to static and dynamic load response, rotordynamic control and response, and reliability and maintainability. The scope of this work includes:

a. The construction and operation of a small rotor model (“mini-mockup”) to develop and verify the analytical tools needed to control a vertical rotor supported by electromagnetic bearings. The rotor is supported on one axial and two radial EMBs and uses rolling element catcher bearings for backup.

b. A scale model of the turbomachine rotor (“RSM”) that will be tested to verify the performance of the EMB control system, including its redundancy and on-line maintainability, and to provide a benchmark for validation of the analytical design tools. The RSM will incorporate a model of the generator rotor and a model of the turbocompressor rotor. These rotors will be joined by a flexible coupling. Each of the rotors will be supported on one axial and two radial EMBs. Additional devices will be installed to simulate magnetic and/or mechanical forces representing those that would be seen by the exciter, generator, compressors and turbine. The RSM will utilize conventional rolling element catcher bearings for both radial and axial backup support since catcher bearing (CB) performance cannot be accurately simulated in a scale model.

c. Sensors to determine the rotor position will be used in conjunction with the EMB support. The testing will be done to evaluate several designs and to determine their response characteristics as a function of physical properties (e.g., inductance, resistance, magnetic permeability) and operating parameters (e.g., frequency, gap width). In addition to providing the basis for sensor selection, these data will provide input to the design of the EMB control system. The scope of these tests includes radial and axial position sensors, plus turning angle sensors that can be used to measure shaft rotation angle and speed.

d. The static capacity and dynamic response of radial EMBs will be tested taking into consideration the design of the control system, the power electronic amplifiers and the physical properties of the EMB stator and rotor. These data will provide input to the sizing of the EMBs and to the design of the EMB control system. These tests will include measurements of non-rotating static and dynamic responses of a typical large radial EMB.

4. Demonstrate the **Catcher Bearings (CB) Subsystem** adequacy to withstand the required duty cycle and rotordynamic control and response characteristics during rundown. The scope for this development work includes:

   a. All of the CB concepts presently being considered involve physical contact between stationary and rotating surfaces during rundown following loss of EMB support. For the reference CB designs, which are gas bearing-plain bearing hybrids, physical contact takes place during a short period of time at the initiation of backup support. The associated tribological issues are compounded by the dry helium environment and restriction on the use of lubricants. For this reason, tribological tests are needed to determine the properties of materials pairs in sliding contact in helium, both with and without prospective lubricants.

   b. Mock-ups of the CB will be used as the initial means of verifying the design concepts of the radial and axial CBs. The CB mock-up test facility will simulate both axial and radial loadings during rotor rundown following a loss of EMB support. Axial loads will be imposed by a pneumatic or electromagnetic loading device. Radial loading will be provided by selected imbalance weights. In order to provide adequate confidence in the results, the tests will be done with a full scale representation of the CBs under simulated full loads.

   c. Testing full-scale radial and axial CB “pilot samples” will be the final CB component development. After testing, the full-scale pilot samples will be installed in the full-scale turbocompressor that will be used in the Integrated TC Test. The scope of pilot sample testing will involve verifying the functionality of basic CB design features at full-scale and at nominal loads.

5. Demonstrate the adequacy of the **Flexible Coupling** between the turbine and the compressor rotors for the required duty cycle. The scope for this development work includes:

a. The flexible coupling will be tested to verify its performance under reactor operating conditions. This test program will use a scale model of the coupling, and it will include three stages. In the first stage, a scale model of the flexible coupling will be tested under static conditions (no rotation), under maximum loads and for several axial, radial and angular shaft displacements. In the second stage, the scale model will be tested under cycling loading without rotation. In the third stage, the scale model will be tested with the shafts rotating under simulated reactor operating conditions and plant design duty cycle. At the end of this test program a full scale prototype of the flexible coupling will be manufactured, mounted on the TC prototype and tested again as part of the full-scale TC integrated test.

6. Demonstrate performance characteristics of the **Recuperator**, including the performance of a heat transfer element, the uniform helium flow distribution at the recuperator module inlets, and the feasibility of manufacturing large number of elements. The scope for this development work includes:

   a. The verification of the performance of a single heat transfer element of the recuperator will be done using two different models. A simple model of the primary heat transfer surface (flat element model) will first be used to optimize the configuration of the fins (intensifier) located in the low-pressure section of the heat exchanger element. Following the selection of the optimum fin configuration, a full-scale model (prototype) of a recuperator heat exchange element will be built and tested. This element will consist of a tubular casing (106 x 3.5 mm tube) containing a primary surface plate-type heat exchange element.

   b. The flow distribution at the recuperator module high- and low pressures inlet sections will be tested using Plexiglas models of these sections. The tests will be done in air at atmospheric conditions. A uniform flow distribution among the numerous heat exchange elements is very important for achieving the required 95% thermal effectiveness. The Plexiglas models will be modified until a satisfactory flow distribution is obtained.

   c. The primary heat transfer surface inside each element must be large enough to avoid a large number of welds and thin enough to be bent in a shape that can be compacted inside the small element volume. Metal sheets with a width of 2200 mm and a thickness of 0.35 mm have been selected. Manufacturing techniques will be developed to mass-produce a large number of defect-free metal plates of these dimensions. Techniques will also be developed to bend each of these plates into the proper shape and to weld two of these plates together along their edges.

7. Demonstrate performance characteristics of the **Precooler/Intercooler/Generator Gas Coolers** including their thermal performance, the potential for flow induced vibrations, the feasibility of manufacturing large number of heat exchange tubes, and capability of detecting and plugging leaking tubes. The scope for this development work includes:

a. The thermal performance of a precooler/intercooler/gas cooler tube bundle has already been tested. Two models have been used. One model had 12 longitudinal fins in each of the heat exchange tubes, and the other had 16 longitudinal fins. The test results did not meet the design requirements and did not support the theoretical calculations indicating the need for improving the design. Several solutions are now under investigation, including some methods for achieving a better and more uniform interface between the fins and the tube surface. As soon as a new design has been established, models will be built and tested in the same OKBM facility as the one use in the preliminary tests.

b. Flow induced vibrations will be tested using a model that simulates a full-scale tube bundle and its support structure. The test model will be loaded on a shaker table and excitations similar to those expected inside the PCU vessel will be applied. Natural frequencies of a tube bundle, the amplitude of its oscillations and the possibility of any damage caused by tube abrasion or fatigue will be determined.

c. Manufacturing techniques will be developed to address the mass production of up to 50,000 heat exchange tubes with a 9-mm OD and 3200-mm long and the associated welding or brazing techniques to provide an effective connection between the tube outer surface and the fins.

d. Several types of instrumentation for detecting helium leaks in the cooling water stream will be investigated. The process and the tools for locating and plugging a leak in a precooler or intercooler module will also be tested. These tests will include the fabrication of a model of a precooler/intercooler module, including the PCU vessel maintenance access header and the area outside this access. This test will use the remote handling tools designed for this purpose.

8. Demonstrate performance characteristics of the **In-Vessel Metalwork[44] and Gas Mixer** including uniform coolant flow distribution and mixing within the PCU integrated system, performance of large bellows in helium, and development of techniques for PCU components inspection, maintenance and replacement. The scope for this development work includes:

   a. The uniformity of the coolant flow will be verified in three areas of the PCU coolant loop: (1) from the turbine outlet to the low-pressure recuperator module outlets, (2) from the precooler modules outlets to the low-pressure compressor inlet, and (3) from the intercooler module outlets to the high-pressure compressor inlet. The test will use Plexiglas models of these areas of the PCU coolant flow path. The tests will be done in air at atmospheric conditions. The Plexiglas models will be modified until satisfactory flow distributions are obtained.

[44] "In-Vessel Metalwork" is a literal translation for in-vessel ducting and other transition pieces providing the flow path between the various components within the PCU pressure vessel.

b. A full-scale model of the middle support bellows will be tested in stagnant helium with a helium pressure inside of 4.7 MPa and a pressure outside of 0.1 MPa. The bellow will be kept at 500 ºC. Once the test conditions are achieved, the bellow will be compressed and cycled until the required number of operation cycles is accumulated. During the cycling process, the bellow will be subjected to several simulations of the pressure drop that occurs when the protection system opens the bypass valves to prevent turbomachine over-speed. After completion of the planned cyclic operation, the bellow will undergo careful inspection (visual inspection, dye penetrant test, etc.) to determine whether continued cycling would lead to the bellow's failure.

c. Access for inspection, maintenance or replacements of components inside the PCU is an issue due to the large number of components packed in a relatively small space and the radioactive contamination of some of these assemblies. This issue will be addressed during testing of the full scale turbocompressor and by detailed evaluation of access areas and remote handling using mock-ups of the PCU-mounted components throughout the PCU final design.

9. Demonstrate the performance of a **Full-Scale Turbocompressor Prototype** in helium under nominal temperatures and volumetric flows in order to verify its operation on EMB over the entire design speed range, and during rundown on catcher bearings. The scope for this development work includes:

   a. The selection of a flexible membrane coupling between the turbocompressor and the generator effectively decoupled these two components. Further, the size of the turbocompressor is such that demonstration of the EMB support system for either component is considered adequate demonstration of both. Therefore, considering the existing experience with vertical hydroelectric generators, it was concluded that this technology demonstration should concentrate only upon the construction and operation of a full-scale prototype of the turbocompressor, which was judged to be the more challenging of the two components. The scope of this development work includes also the fabrication of a test facility that will simulate the GT-MHR operating conditions for the turbocompressor prototype.

6.4.2 Technology for 950 ^{o}C Operation

The technology development plan for the 850 ^{o}C baseline PCS design was described in Section 6.4.1. In order to increase the inlet temperature to 950 ^{o}C, some additional development will be required. Fortunately, increasing inlet temperature to PCS will have little or no impact on compressors, pre- and inter-coolers, pressure vessel and electromagnetic bearings. However, the following components will be affected and thus will require further technology development.

1. Hot-Gas Duct (HGD)[45]

The reference design is based on a maximum helium temperature of 850 °C. High temperature insulation (HTI) is used inside the HGD to keep the duct temperature < 600 °C. The resistance and thickness of the HTI will have to be upgraded.

2. Turbine blades

The turbine blade materials do not need cooling to achieve the desired life of 60,000 hours at a helium inlet temperature of 850 °C. However, the lifetime of turbine blades will be significantly reduced if maximum temperature exceeds 850 °C. Hence, development will be required to qualify a blade cooling design and to find suitable thermal barrier coatings. The mechanical design of He turbine blades is significantly different from that of combustion turbine blades, and the design of the blade cooling system will be different. Consequently, experimental confirmation of the blade cooling system is judged to be necessary.

In addition to blade cooling, a thermal-barrier coating will be necessary on the first few stages because of the efficient heat transfer from the helium. Thermal-barrier coatings are routinely used in combustion turbines; however, they evidently have not been used on He turbines (Gandy 2001). The optimal coating composition and application technique will be determined, and the coating stability and effectiveness will be confirmed in high temperature, high velocity He. If coating were to spall off prematurely, blades could overheat and fail. Moreover, spallation fragments could be transported into core and become activated, generating a radioactive aerosol in the primary circuit; consequently, the selected coating will have a low Co content.

3. Recuperator

Higher inlet temperature to the turbocompressor will increase the outlet temperature from TC and inlet temperature to the recuperator from current design value of 510 °C to ~ 580 °C. Thus, creep life of the recuperator material of construction will be determined. Redesign of the support system for higher thermal stresses associated with higher temperature differences may be required.

4. Internal PCS Ducting

Ducting and other mechanical connections within the PCS are used to minimize the internal leakages from different parts of the vessel. Mockup tests will be performed to determine the effect of higher temperature on the baseline design.

6.5 Design Verification and Support Program

The current NGNP and NHI technology development programs are largely generic because there is no reference NGNP design. In fact, many fundamental design selections have yet to be made: reactor core type, IHX configuration, hydrogen production process, etc. Consequently, it is not surprising that the current TDPs do not address DV&S DDNs to a significant degree.

[45] The HGD is actually part of the Reactor Internals System but is included here for convenience.

When the reference NGNP design is chosen, additional TDPs will need to be prepared that will address the DV&S DDNs for key SSCs. It is expected that new design-specific TDPs will include plans for the Reactor System, Vessel System, Reactor Cavity Cooling Systems, etc.; these anticipated new TDPs are included in Table 6-1.

6.6 Hydrogen Production Programs

In the USA, nuclear hydrogen production technologies are being developed under the DOE Office of Nuclear Energy, Science, and Technology (DOE/NE) Nuclear Hydrogen Initiative. At present, technology development plans have been developed only at a high level. These plans are described in the Nuclear Hydrogen Initiative Ten Year Program Plan (NHI Plan 2005). As discussed in the NGNP PMPP (2006), the NHI plan is generally consistent with the NGNP construction schedule. The NHI plan covers both thermochemical water splitting and HTE (based on planar-cell technology). More detailed technology development plans for the SI and HTE processes should be developed during the Conceptual Design phase to ensure that the DDNs will be satisfied.

6.6.1 Sulfur-Iodine Thermochemical Water-Splitting

A basic technology development schedule for hydrogen production by the Sulfur-Iodine cycle is shown in Figure 6-5.

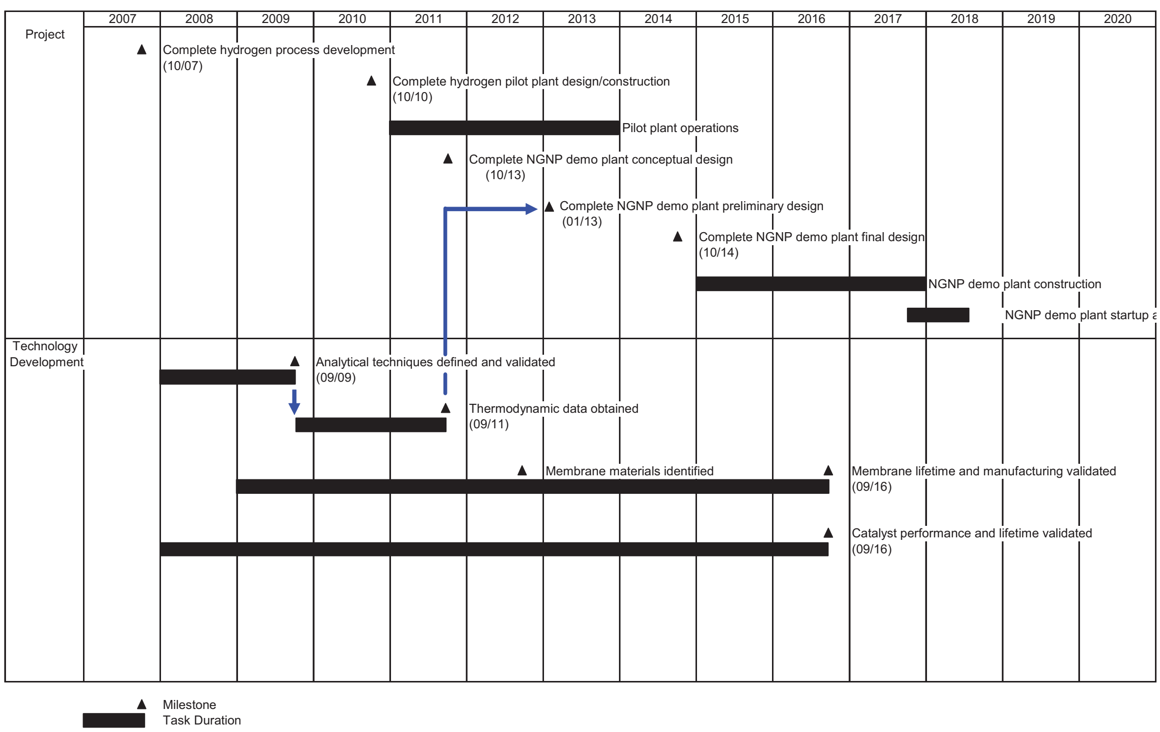

Figure 6-5. Basic Technology Development Plan for SI Process

For large-scale hydrogen production, efficient design of process equipment is important. Unit operations typical in the chemical process industries are used in the SI thermochemical process. Design methods for distillation columns, heat exchangers, etc., at plant scale are mature and well understood. This circumstance is to the advantage of the SI process, as development of new unit operations is not required as the process is scaled up. The fundamental chemical processes of the SI cycle have been demonstrated, including closed-loop operation conducted in Japan. The DDNs for the SI process outlined in Section 5 focus primarily on efforts to reduce uncertainties in equipment design and materials selection. These uncertainties lead to safety factors and overdesign which can increase capital costs. Thermal efficiency can also suffer in this case if heat exchanger wall thicknesses are oversized for uncertain pressure or corrosion considerations rather than for efficient heat transfer.

6.6.2 High Temperature Electrolysis

As shown in Figure 6-6, Toshiba has developed a preliminary integrated schedule for HTE design and technology development based on their tubular-cell concept. This schedule is compliant with the NGNP schedule. Toshiba has also developed a high-level technology-development matrix (see Table 6-3) as the technology development evolves from cells, to units consisting of multiple cells, to pilot-scale modules consisting of multiple units, to engineering-scale modules, and finally to the NGNP demonstration consisting of 10 engineering-scale modules.

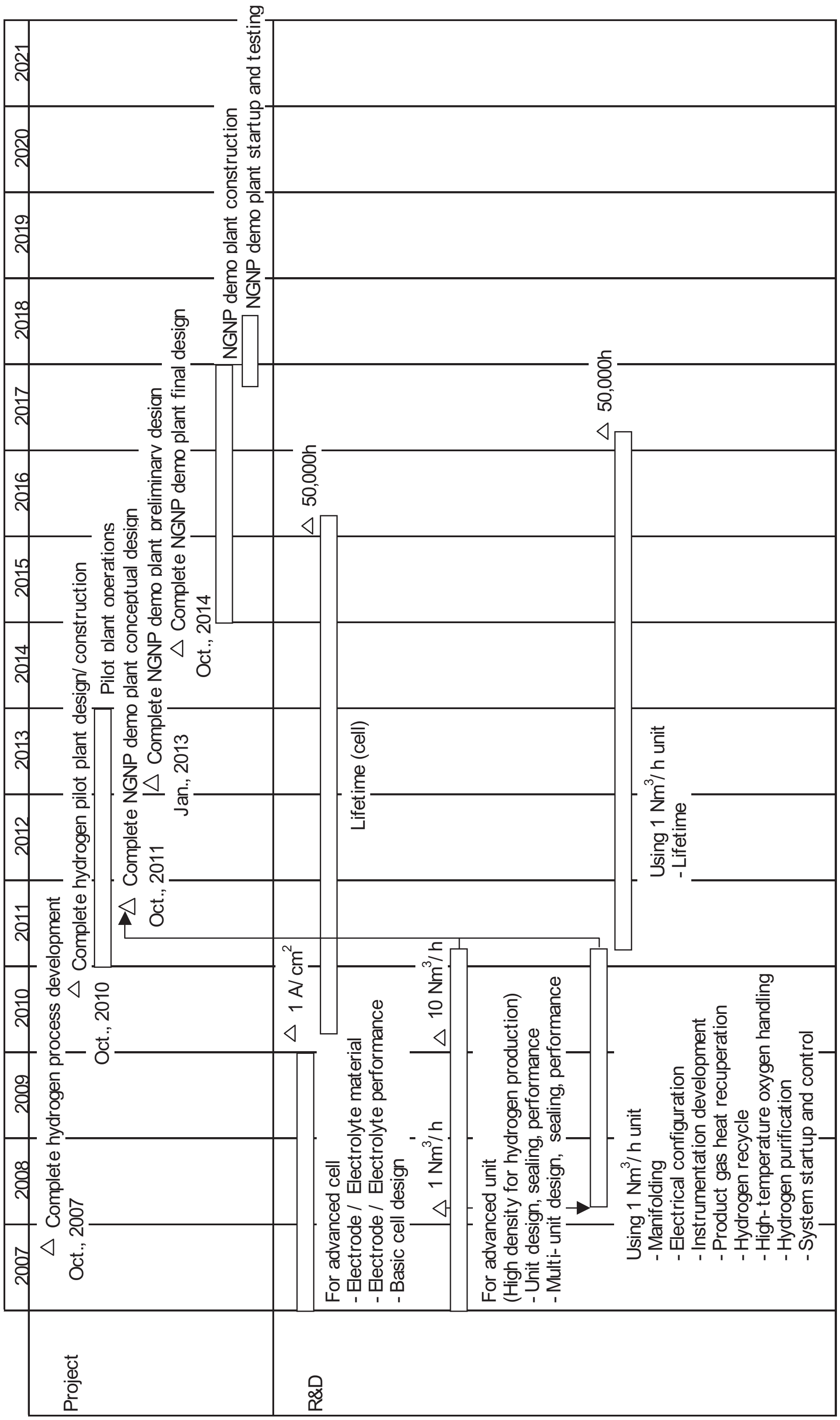

Figure 6-6. Toshiba Integrated Plan for HTE Design and Technology Development

6.7 Design Methods Development and Validation Program

An extensive code development and validation program is presented in the NGNP Design Methods Development and Validation Research and Development Program Plan (Schultz 2004). The Plan is portrayed as being supportive of both prismatic and pebble-bed core designs. The emphasis is heavily upon core nuclear and thermal/fluid flow computational methods. Design methods for predicting coated-particle fuel performance and fission product transport are not addressed; the Plan states that AGR Fuel Program will provide the necessary design methods for those applications. While the AGR Plan does include development of improved component models, etc., it does not include scope for developing advanced computational tools for full-core performance analysis or for predicting RN transport throughout the plant, and tritium transport is not addressed at all.

From GA team's perspective, the emphasis in this NGNP methods development plan is misguided. At least for prismatic MHRs, the currently available computational tools for core nuclear analysis and thermal/fluid flow analysis are adequate for NGNP Conceptual and Preliminary Design and, perhaps, for Final Design as well. The traditional GA design methods for analyzing prismatic HTGRs, that were first developed to support the design and licensing of Fort St. Vrain and the large HTGRs in the 1970s, are still available. However, for nuclear analysis, the traditional codes have been largely supplanted at GA by industry standard codes, such as DIF3D and MCNP, and for thermal, flow, and structural analyses, commercial codes, such as ANSYS, RELAP5, SINDA/FLUENT, and CFX, are already being used routinely by the GA team. In contrast, the design methods for predicting fuel performance and fission product transport are in need of modernization and upgrading to support NGNP design and licensing.

On previous DOE HTGR programs, design methods development and validation have been design tasks rather than technology tasks because the computational tools required for a particular application and the required predictive accuracies are often design specific. In addition, there is often a trade off to be made between development and validation of highly sophisticated computational tools and the addition of design margin to accommodate calculational uncertainties that may be introduced by the use of less complex design methods.

As discussed in Section 4.2.1.2.1 in the context of RN source term predictions, all design methods do not *a priori* have to be highly accurate; however, there must be sufficient design margin to reliably account for the uncertainties in the predictions. In some cases, it has proven impractical or uneconomical to add large design margins; consequently, the design methods for such applications are required to be highly accurate. Once the NGNP Conceptual Design is established, trade studies can be performed to determine the required predictive accuracies of the various design methods. Such trade studies will allow the methods development tasks to be focused and prioritized.

Even before such NGNP-specific trade studies are available, there is sufficient international experience with the design of prismatic HTGRs to conclude that some of the planned workscope in the NGNP methods development plan is unnecessary and would not be a wise expenditure of finite resources (both human and financial). For example, the Plan includes the construction of a large-scale mockup of the Reactor System (termed "Integral VHTR" in the PPMP) for validation of thermal/fluid flow predictive methods for normal operation and accidents. This $40M facility is not needed for a prismatic-core Reactor System. Some small-scale mockup tests might be prudent (e.g., to characterize hot streaks in the hot duct), but such DV&S tests would in general be highly design specific, and their need cannot be determined until late in the Preliminary Design phase. Moreover, such integral thermal/flow data will be generated during the hot-flow tests (prior to initial criticality) during the reactor startup testing phase.

The Plan includes other DV&S tests as well, including an RCCS performance test. To reiterate, DV&S tests are typically design specific, and the test requirements cannot be reliably defined until well into Preliminary Design. Such tests should be included in a series of stand-alone DV&S TDPs rather than in a generic methods development plan.

6.8 Spent Fuel Disposal Program

As described in Section 5.2.7, the preferred option for MHR spent fuel disposition is the direct disposal of unprocessed spent fuel elements in a geologic repository (e.g., Richards 2002). The spent-fuel disposal DDNs for the commercial GT-MHR (Hanson 2002) were listed in Table 5-1; the corresponding DDNs for the NGNP should be virtually identical. A confirmatory test and analysis plan defining experimental programs to satisfy these DDNs has been prepared for the commercial GT-MHR (Hanson 2002).

The tests proposed in the plan are summarized Table 6-4. The tests would be relatively inexpensive compared to the high contemporaneous costs of performing nuclear fuel R&D. The experiments would be laboratory scale and would typically require test facilities, instrumentation, and support services found in a modern, well-equipped radiochemistry laboratory; some tests may require hot cells, depending upon sample size and radiation levels, especially for sample recovery and preparation. The experiments would be technically simple but generally of long-term duration (in some cases, multiple years). The most sophisticated apparatus would likely be conventional high-pressure autoclaves for accelerated corrosion and leaching tests with (simulated) liquid groundwater. Certain of the test programs could begin early, essentially immediately upon availability of funding, because they would utilize existing samples of unirradiated and irradiated TRISO-coated fuel particles and H-451 graphite which are available in quantity at ORNL.

None of the R&D proposed in (Hanson 2002) has been funded to date. Internationally, the French have described R&D programs to support the disposition of spent MHR fuel (Roudil 2006) which would address some of the DDNs listed in Table 6-4.

6.9 References for Section 6

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Hanson, D. L., and M. B. Richards, "[Commercial GT-MHR] Spent Fuel Disposal Confirmatory Test and Analysis Plan," PC-000503, Rev. 0, General Atomics, June 2002.

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[NGNP Materials R&D Plan] "Next Generation Nuclear Plant Materials Research and Development Program Plan," INEEL/EXT-05-00758, Rev. 2, Idaho National Laboratory, September 2005.

[NHI Plan] "Nuclear Hydrogen Initiative Ten Year Program Plan," US Department of Energy, March 2005.

[PCU TDP] PCU "Technology Demonstration Program Plan, Revision 2005 (Draft)," Product No: 08.03-006.01A, OKBM, 2005 *(Business Confidential).*

[PPMP] Weaver, K., et al., "Next Generation Nuclear Plant Project Preliminary Project Management Plan," INL/EXT-05-00952, Rev. 1, Idaho National Laboratory, March 2006.

Richards, M., "Assessment of GT-MHR Spent Fuel Characteristics and Repository Performance," GA Document PC-000502, Rev. 0, General Atomics, March 2002.

Roudil, D., et al., "Long Term Behavior Studies of HTR Fuel Particles under Disposal Conditions: Issues, Program and Initial Results," in <u>Proceedings HTR2006: 3rd International Topical Meeting on High Temperature Reactor Technology, October 1-4, 2006, Johannesburg, South Africa</u>.

Schultz, R. R., et al., "NGNP Design Methods Development and Validation Research and Development Program Plan," INEEL/EXT-04-02293, Rev. 0, Idaho Engineering and Environmental Laboratory, September 2004.

Table 6-1. Evaluation of Current Technology Programs

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program (AGR Plan/1 2005)				
C.07.01	**Fuel Process Dev.**			
N.07.01.06	Mass Production of High Quality UCO TRISO Fuel	Demonstration of economical mass-production of coated-particle fuel is not within the scope of the AGR Fuel Program nor is NGNP fuel supply addressed in the NGNP PPMP	Demonstration of a viable fuel supply is essential to deployment of commercial MHRs in the USA	The NGNP Project should build, license, and operate a fuel manufacturing pilot plant for NGNP to demonstrate the viability of economical mass production of coated-particle fuel, thereby satisfying the fuel fabrication process DDNs identified in Table 5-1
N.07.01.07	As-manufactured Quality of LEU UO_2 (NFI extended burnup fuel)	Described in NGNP PPMP, but not incorporated into AGR Fuel Program. Results of recent NGNP fuel acquisition study performed by AGR Fuel Program implies that this option is no longer being considered	GA views NFI as the only viable source of fuel for the NGNP initial core with a 2018 startup. NFI must demonstrate that it can mass produce fuel to NGNP quality requirements. The existing irradiation performance database for the NFI extended burnup fuel is limited and likely inadequate to support NGNP licensing	Contract with NFI to make proof test fuel for irradiation in ATR. Expand AGR Fuel Program to include irradiation testing of NFI fuel
N.07.02	**Fuel Materials**			
N.07.02.08	Irradiation Performance of LEU UO_2 (NFI)	Described in NGNP PPMP, but not incorporated into AGR Fuel Program. Results of recent NGNP fuel acquisition study performed by AGR Fuel Program suggests that this approach is no longer being considered	GA views NFI as the only viable source of fuel for the NGNP initial core with a 2018 startup. NFI must demonstrate that it can mass producer fuel to NGNP quality requirements. The existing irradiation performance database for the NFI extended burnup fuel is limited and likely inadequate	Contract with NFI to make proof test fuel for irradiation in ATR. Expand AGR Fuel Program to include irradiation testing of NFI fuel

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
			to support NGNP licensing	
N.07.02.09	Accident Performance of LEU UO_2 (NFI)	Described in NGNP PPMP, but not incorporated into AGR Fuel Program. Results of recent NGNP fuel acquisition study performed by AGR Fuel Program imply that this approach is no longer being considered.	The accident condition fuel performance database for the NFI extended burnup fuel is limited and is likely inadequate to support NGNP licensing	Contract with NFI to make proof test fuel for irradiation in ATR. Expand AGR Fuel Program to include irradiation testing and safety testing of NFI fuel
N.07.03	**Radionuclide Transport**			
C.07.03.04	Fission Product Diffusivities/Sorptivities in Graphite	Geometry of AGR-3 and AGR-4 test trains is unsuitable for characterizing fission metal transport in graphite. Measurement of FP sorptivities for fuel-compact matrix and fuel-element graphite not included in AGR Plan/1.	Fuel-element graphite is major barrier to release of fission metals and actinides from the core during normal operation and core heatup accidents.	Modify AGR-3/-4 capsule geometry (Hanson 2005b). Add sorptivity measurements (Hanson 2006b) to R&D program.
C.07.03.05	Tritium Permeation in Heat Exchanger Tubes	Not included in AGR Plan/1	H-3 can permeate through heat exchangers and contaminate the product hydrogen.	Add H-3 permeation measurements through HX materials (Hanson 2006a) to AGR program.
C.07.03.06	Tritium Transport in Core Materials	Not included in AGR Plan/1	Large quantities of H-3 are produced in core but are evidently largely retained. Core graphite is major sink for H-3.	Add measurements of H-3 release from intact TRISO particles and control materials and H-3-on-graphite sorption measurements (Hanson 2006a) to AGR program.
C.07.03.07	RN Deposition Characteristics of Structural Materials	Nominally included in AGR Plan/1 but no testing until FY12.	Large uncertainties in dose rates from plateout on turbine complicate O&M. Major issue for direct-cycle PCS designs	Accelerate sorptivity measurements (Hanson 2006b) and construction of out-of-pile loop (Hanson 2005a)
C.07.03.08	Decontamination Protocols for Turbine Alloys	Nominally included in AGR Plan/1 but no testing until FY12.	Effective decontamination of turbine would greatly simplify maintenance (e.g., reblading).	Accelerate sorptivity measurements (Hanson 2006b) and construction of out-of-pile loop (Hanson 2005a) which will generate samples for decontamination testing.
C.07.03.09	RN Reentrainment	Nominally included in AGR Plan/1 but no	Liftoff of plateout activity	Accelerate construction of out-

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
	Characteristics for Dry Depressurization	testing until FY12.	during rapid depressurization accidents is major source term for MHR with VLPC.	of-pile loop (Hanson 2005a)
C.07.03.10	RN Removal Characteristics for Wet Depressurization	Nominally included in AGR Plan/1 but no testing until FY12.	Washoff of plateout activity during H_2O ingress accidents can be major source term for MHR with VLPC if pressure relief occurs.	Add H_2O injection capability to out-of-pile loop (Hanson 2005a) but defer testing until significance of H_2O ingress to NGNP safety case is known.
C.07.03.11	Characterization of the Effects of Dust on RN Transport	Nominally included in AGR Plan/1 but no testing until FY12.	Dust in primary circuit can alter plateout distribution and increase liftoff during rapid depressurization accidents.	Add dust injection capability to out-of-pile loop (Hanson 2005a).
C.07.03.14	Fission Gas Release Validation Data	In-pile loop needed to obtain release data at high pressure. Construction of in-pile loop included in AGR Plan/1, but no serious feasibility study conducted. Loop design scheduled for FY13.	Design methods for predicting source terms need to be validated before NRC will grant Operating License. Iodine is dominant radionuclide for off-site doses.	Determine technical feasibility and cost and schedule for constructing in-pile loop in ATR (2004b).
C.07.03.15	Fission Metal Release Validation Data	In-pile loop needed to obtain release data at high pressure and high flow rates. Construction of in-pile loop included in AGR Plan/1, but no serious feasibility study conducted. Loop design scheduled for FY13.	Design methods for predicting source terms need to be validated before NRC will grant Operating License.	Determine technical feasibility and cost and schedule for constructing in-pile loop in ATR (2004b).
C.07.03.16	Plateout Distribution Validation Data	In-pile loop needed. Construction of in-pile loop included in AGR Plan/1, but no serious feasibility study conducted. Loop design scheduled for FY13.	Design methods for predicting source terms need to be validated before NRC will grant Operating License. Plateout on turbine major issue for direct-cycle PCS designs.	Determine technical feasibility and cost and schedule for constructing in-pile loop in ATR (2004b).
C.07.03.17	Radionuclide "Liftoff" Validation Data	In-pile loop needed. Construction of in-pile loop included in AGR Plan/1, but no serious feasibility study conducted. Loop design scheduled for FY13.	Design methods for predicting source terms need to be validated before NRC will grant Operating License. Liftoff of plateout activity	Determine technical feasibility and cost and schedule for constructing in-pile loop in ATR (2004b).

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
			during rapid depressurization accidents is major source term with VLPC.	
C.07.03.18	Radionuclide "Washoff" Validation Data	In-pile loop needed. Construction of in-pile loop included in AGR Plan/1, but no serious feasibility study conducted. Loop design scheduled for FY13.	Design methods for predicting source terms need to be validated before NRC will grant Operating License. Washoff of plateout activity during H_2O ingress accidents can be major source term with VLPC if pressure relief occurs.	Determine technical feasibility and cost and schedule for constructing in-pile loop in ATR (2004b).
N.07.03.19	Physical and Chemical Forms of RNs Released during Core Heatup	AGR Plan/1 calls for thermochemical analyses and "…small-scale tests if necessary…" to determine chemical forms of radionuclides released from the core during core heatup accidents.	The transport behavior of key RNs in (I, Sr, Cs, Te, and Ag) in the VLPC cannot be determined without knowing their physical form and chemical composition. Iodine is the highest priority.	Define R&D program (Hanson 2007) and add to AGR Program. In particular, the trapping systems for the postirradiation heating furnaces should be modified such that the chemical and physical forms of the released RNs can be determined.
N.07.03.20	RN Sorptivities of VLPC Surfaces	Not included AGR Plan/1	Molecular deposition is expected to be the dominant removal mechanism for key RNs, including iodines, in the VLPC.	Define R&D program (Hanson 2007) and add to AGR Program.
N.07.03.21	Qualification of Coatings with High Iodine Sorptivity	Not included AGR Plan/1	Molecular deposition is expected to be the dominant removal mechanism for key RNs, including iodines, in the VLPC. Highly sorptive coatings or paints may increase iodine retention.	Define R&D program (Hanson 2007) and add to AGR Program.
N.07.03.22	Validation Data for Predicting RN Transport in VLPC	Not included AGR Plan/1	LWR test experience (e.g., the PHEBUS tests) has demonstrated that a representative source of	Define R&D program (Hanson 2007) and add to AGR Program. Test facility may be combined with in-pile loop.

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
			radionuclides (i.e., irradiated fuel) is essential for containment response tests. The irradiated fuel must contain a sufficient quantity of I-131.	
[Commercial GT MHR] Spent Fuel Disposal Confirmatory Test and Analysis Plan (Hanson 2002)				
N.07.05	**Spent Fuel Disposal**			
N.07.05.01 - N.07.05.14		Plan judged adequate at preconceptual design phase. No work funded and no announced plans for future funding.		
Reactor System DV&S Plan (TBD)				
C.11.01.01 - C.11.04.06	Reactor System	No Reactor System DV&S plan has been written to date. Most of the RS materials DDNs are addressed in the NGNP Materials R&D Plan (2005).	Reactor System design must be verified to meet top-level requirements.	Prepare Reactor System DV&S plan during Preliminary Design
Vessel System DV&S Plan (TBD)				
C.12.01.01 - C.12.01.04	Vessel System	No Vessel System DV&S plan has been written to date. Most of the VS materials DDNs are addressed in the NGNP Materials R&D Plan (2005)	Vessel System design must be verified to meet top-level requirements. VS part of primary pressure boundary.	Prepare Vessel System DV&S plan during Preliminary Design
Energy Transfer Technology Development Plan (TBD); NGNP R&D Plan (2005); NGNP Materials R&D Plan (2005)				
N.13.01	**PHTS Circulator**			
N.13.01.01 - N.13.01.03		No Primary Helium Circulator DV&S plan has been written to date.	PHC design must be verified to meet top-level requirements.	Add PHC to Energy Transfer TDP.
N.13.02	**Intermediate Heat Exchanger (IHX)**			
N.13.02.01 - N.13.02.09		No IHX DV&S plan has been written to date. IHX materials qualification included in Material R&D Plan (2005), and some IHX DV&S tests included in Methods Development Plan (2005).	IHX design must be verified to meet top-level requirements.	Add IHX to Energy Transfer TDP.

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
C.14.01.01 - C.11.01.04	**SCS Circulator**	No SCS DV&S plan has been written to date.	SCS design must be verified to meet top-level requirements, including investment protection.	Add SCS to Energy Transfer TDP.
C.14.04.01 - C.14.04.10	**SCS Heat Exchanger**	No SCS DV&S plan has been written to date.	SCS design must be verified to meet top-level requirements, including investment protection.	Add SCS to Energy Transfer TDP.
Reactor Cavity Cooling System DV&S Plan (TBD)				
C.16.00.01 - C.16.00.06	Reactor Cavity Cooling System	No Reactor Cavity Cooling System DV&S plan has been written to date. Some RCCS DV&S tests included in Methods Development Plan (2005).	RCCS design must be verified to meet top-level requirements.	Prepare RCCS DV&S plan during Preliminary Design
Fuel Handling and System DV&S Plan (TBD)				
C.21.01.01 - C.21.01.09	Fuel Handling and Storage System	No Fuel Handling and Storage System DV&S plan has been written to date.	FH&SS design must be verified to meet top-level requirements.	Prepare FH&SS DV&S plan during Preliminary Design
Reactor Protection DV&S Plan (TBD)				
C.31.01.01 - C.31.01.02	Reactor Protection System	No Reactor Protection System DV&S plan has been written to date.	RPS design must be verified to meet top-level requirements.	Prepare RPS DV&S plan during Preliminary Design
Plant Control, Data and Instrumentation System DV&S Plan (TBD)				
C.34.01.01 - C.34.01.09	Plant Control, Data and Instrumentation System	No Plant Control, Data and Instrumentation System DV&S plan has been written to date.	PCD&IS design must be verified to meet top-level requirements.	Prepare PCD&IS DV&S plan during Preliminary Design
PCU TDP (2005); NGNP R&D Plan (2005); NGNP Materials R&D Plan (2005)				
RF DDNs for PCU (Table 5-7)	PCU DV&S	OKBM PCU TDP judged capable of qualifying baseline PCU design for operation at 850 ^{o}C.	PCU is high-risk FOAK machine and mission critical.	Carefully monitor progress of PCU technology demonstration program at OKBM. Revise and supplement TDP as necessary.
N.41.01.01 -	Modification of OKBM	Incremental technology development for	PCU is high-risk FOAK	Prepare an incremental TDP to

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
N.41.04.01	PCU Design for 950 °C Operation	950 °C operation not addressed in any existing TDP.	machine and mission critical.	supplement the PCU TDP for 950 °C operation.
N.41.01.01	Service Lifetime of Uncooled Turbine Blades at 950 oC	Turbine blade alloys not included in NGNP Materials R&D Plan (2005)	Qualification of turbine alloys (e.g., IN 100) high-priority materials DDNs	Add leading candidate turbine blade alloys (IN 100, IN738) to materials R&D program.
[Component DV&S TDPs][46]				
N.XX.YY.ZZ		As the NGNP design matures, it is anticipated that new DV&S DDNs will be identified for various SSCs		Prepare additional DV&S plans during Preliminary Design as required.
NGNP R&D Plan (2005); NGNP Materials R&D Plan (2005); [Energy Transfer Technology Development Plan (TBD)]				
N.42.02	**Isolation Valves**			
N.42.02.01		No DV&S plan has been written to date which addresses the high temperature isolation valves of the secondary system	HTIV design must be verified to meet top-level requirements.	Add HTIV to Energy Transfer TDP.
NHI Plan (2005) – SI Process				
N.44.01	**[Sulfuric Acid Decomposition]**			
N.44.01.01	Catalyst Performance	DOE high-level plan is generally responsive	Better understanding of actual plant capacity factors and capital/operating costs	Prepare detailed SI TDP
N.44.02	**[Bunsen Reaction]**			
N.44.02.01	Reactor Design	DOE high-level plan is generally responsive	Optimized reactor design, reduced cost, reduced side reaction	Prepare detailed SI TDP
N.44.02.02	Refined Thermodynamic Model	DOE high-level plan is generally responsive	Improved design reliability	Prepare detailed SI TDP
N.44.03	**[Hydrogen Iodide Decomposition]**			

[46] No TDP for component DV&S has been for the NGNP program. The Engineering Development Plan for the NP-MHTGR provides an indication of the scope of such a plan. Practically, preparation of component DV&S TDP during preconceptual design is probably premature.

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
N.44.03.01	HI/H_2 Membrane Separation	DOE high-level plan is generally responsive	Reduced cost compared to refrigerated phase separation	Prepare detailed SI TDP
N.44.03.02	Refined Thermodynamic Model	DOE high-level plan is generally responsive	Improved design reliability	Prepare detailed SI TDP
N.44.03.03	Liquid HI Decomposition	DOE high-level plan is generally responsive	-Higher process efficiency -Lower capital cost -High pressure (50 bar) H_2	Prepare detailed SI TDP
N.44.04	**<u>Materials Compatibility</u>**			
N.44.04.01	Corrosion performance	DOE high-level plan is generally responsive	Optimal design, reduced cost	Prepare detailed SI TDP
N.44.04.02	Equipment Manufacturability	DOE high-level plan is generally responsive	Optimal design, reduced cost	Prepare detailed SI TDP
NHI Plan (2005) – HTE Process				
N.45.01	<u>SOE Cells</u>			
N.45.01.01	Electrode / Electrolyte Materials	Toshiba high-level plan is generally responsive	Data needed to support SOE cell design and model SOE cell performance	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.01.02	SOEC Design and Performance	Toshiba high-level plan is generally responsive	Data needed to support SOE unit design and model SOE unit performance	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.02	**<u>SOE Units</u>**			
N.45.02.01	SOE Unit Design and Performance	Toshiba high-level plan is generally responsive	Data needed to support SOE pilot-scale module design and model SOE pilot-scale module performance	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.02.02	SOE Multi-Unit Integration and	Toshiba high-level plan is generally responsive	Data needed to support SOE pilot-scale module design and	Prepare detailed HTE TDP that includes both tubular-cell and

DDN No.	DDN Title	Responsiveness of Technology Plans to DDNs	Data Significance	Recommended Disposition
	Performance		model SOE pilot-scale module performance	planar-cell technology development through pilot plant testing.
N.45.03	**SOE Modules**			
N.45.03.01	SOE Pilot-Scale Module Demonstration	Toshiba high-level plan is generally responsive	Data needed to support SOE engineering-scale module design and model SOE engineering-scale module performance	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.03.02	SOE Engineering Scale Module Demonstration	Toshiba high-level plan is generally responsive	Initial demonstration of a prototype module for commercialization	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.03.03	NGNP SOE Multi-Module Demonstration	Toshiba high-level plan is generally responsive	Provides basis for commercialization of nuclear hydrogen production using HTE technology.	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.04	**HTE Plant Supporting Equipment**			
N.45.04.01	HTE Steam Generator/ Superheater	Toshiba high-level plan is generally responsive	Needed to support HTE plant design.	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.
N.45.04.02	HTE Heat Exchangers	Toshiba high-level plan is generally responsive	Needed to support HTE plant design.	Prepare detailed HTE TDP that includes both tubular-cell and planar-cell technology development through pilot plant testing.

Table 6-2. AGR Fuel Irradiation Tests

Capsule	Test Description	Test Objective/Expected Results
AGR-1	**<u>Shakedown Test/Early Fuel</u>** Contents to include compacts made from early small-coater particles, possible compacts made from German particles, as well as possible unbonded particles and material samples. (Actual AGR-1 test contains only UCO compacts.)	Gain experience with multi-cell capsule design, fabrication, and operation and reduce chances of capsule or cell failures in subsequent capsules, early data on irradiated fuel performance, support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance.
AGR-2	**<u>Performance Test Fuel</u>** Contents to include compacts containing particles made in large coater from key variants in coated particles (e.g., IPyC permeability, anisotropy, uranium dispersion in buffer, continuous vs. interrupted coating), possibly fuel performance modeling material samples, common cell temperatures in four central cells	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE for key fuel product/process variants to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance
AGR-3	**<u>Fission Product Transport</u>** Contents to include compacts of LEU UCO and NUCO particles seeded with designed-to-fail (DTF) fuel LEU UCO and NUCO separately), unbonded kernels, DTF particles	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE. Data on fission gas release from failed particles, fission metal diffusion in kernels, and gas and metal diffusion in coatings for use in development of FP transport models.
AGR-4	**<u>Fission Product Transport</u>** Contents to include DTF driver fuel and specialized samples of compact matrix and graphites	Provide data on fission product diffusivities and sorptivities in compact matrix and graphite materials for use in development of fission product transport models
AGR-5	**<u>Fuel Qualification</u>** Contents to include a single fuel type made using process conditions and product parameters considered to provide best prospects for successful performance based on process development results and available data from AGR-1 and AGR-2, variations in cell irradiation temperatures per test specification	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE in sufficient quantity to demonstrate compliance with statistical performance requirements under normal operation and accident conditions
AGR-6	**<u>Fuel Qualification</u>** Contents to include same fuel type as used in AGR-5, variations in cell irradiation temperatures per test	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE sufficient quantity to demonstrate compliance with statistical performance requirements under normal

Capsule	Test Description	Test Objective/Expected Results
	specification	operation and accident conditions
AGR-7	**Fuel Performance Model Validation** Contents to include same fuel type as used in AGR-5. The irradiation would cycle the fuel thermally and be designed so that some measurable level of fuel failure would occur (i.e., margin test)	Provide irradiation fuel performance data and irradiated fuel samples for safety testing and PIE in sufficient quantity to validate the fuel performance codes and models and to demonstrate capability of fuel to withstand conditions beyond AGR-5 and -6 in support of plant design and licensing.
AGR-8	**FP Transport Model Validation** Contents to include compacts seeded with LEU UCO and NUCO particles with missing buffers, unbonded reference particles, different temperatures among cells, and to include temperature cycling	Provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to determine material properties and fission product gas and metal releases from compacts with known quantities of failed particles for use in validation of fuel performance modeling and fission product transport codes

Table 6-3. Toshiba HTE Technology Development Matrix

<table>
<tr><th></th><th colspan="2">Cell</th><th>Unit</th><th>Multi-Unit</th><th colspan="3">Facility</th></tr>
<tr><th></th><th>Present
0.03 kW
0.01 Nm^3/h</th><th>Advanced
0.1 kW
0.03 Nm^3/h</th><th>Bench
3.1 kW
1 Nm^3/h</th><th>Integrated
31 kW
10 Nm^3/h</th><th>Pilot
190 kW
60 Nm^3/h</th><th>Engineering Demo
1.9 MW
600 Nm^3/h</th><th>NGNP Pilot
19 MW
6000 Nm^3/h</th></tr>
<tr><td>Electrode / Electrolyte material</td><td colspan="2">x</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Electrode / Electrolyte performance</td><td colspan="2">x</td><td>x</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Basic cell design</td><td colspan="2">x</td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Unit design</td><td></td><td></td><td>x</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Unit sealing</td><td></td><td></td><td>x</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Unit performance</td><td></td><td></td><td>x</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Multi-unit design</td><td></td><td></td><td></td><td>x</td><td></td><td></td><td></td></tr>
<tr><td>Multi-unit sealing</td><td></td><td></td><td></td><td>x</td><td>x</td><td></td><td></td></tr>
<tr><td>Multi-unit performance</td><td></td><td></td><td></td><td>x</td><td>x</td><td></td><td></td></tr>
<tr><td>Manifolding</td><td></td><td></td><td>x</td><td>x</td><td>x</td><td>x</td><td></td></tr>
<tr><td>Electrical configuration</td><td></td><td></td><td></td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Instrumentation development</td><td colspan="2">x</td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Heating of feedstock</td><td></td><td></td><td></td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Product gas heat recuperation</td><td></td><td></td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Hydrogen recycle</td><td></td><td></td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>High-temperature oxygen handling</td><td></td><td></td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Lifetime</td><td colspan="2">x</td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Hydrogen purification</td><td></td><td></td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>System startup and control</td><td></td><td></td><td>x</td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>System maintenance</td><td></td><td></td><td></td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>High-pressure operation</td><td></td><td></td><td></td><td></td><td>x</td><td>x</td><td></td></tr>
<tr><td>Hydrogen storage</td><td></td><td></td><td></td><td></td><td></td><td>x</td><td></td></tr>
<tr><td>Demonstration of large-scale hydrogen production</td><td></td><td></td><td></td><td></td><td></td><td>x</td><td>x</td></tr>
</table>

Table 6-4. Tests to Support GT-MHR Spent Fuel Disposal

Design Data Need	Generic Test Description
Long-term mechanical integrity of stressed TRISO coatings	PIE of historical irradiated TRISO particles, TBD
PyC coating oxidation rates (air)	Accelerated (temperature) tests with TRISO particles
SiC coating oxidation rates (air)	Accelerated tests with TRISO particles (OPyC removed)
Matrix oxidation rates (air)	Accelerated lab tests
H-451 graphite oxidation rates (air)	Accelerated lab tests with unirradiated/irradiated samples
Graphite noncombustibility demonstration	Combustibility testing per ASTM standard (or equivalent)
PyC coating corrosion rates (groundwater)	Accelerated lab tests with spectrum of water chemistries
SiC coating corrosion rates (groundwater)	Accelerated lab tests with spectrum of water chemistries
Matrix corrosion rates (groundwater)	Accelerated lab tests with spectrum of water chemistries
H-451 graphite corrosion rates (groundwater)	Accelerated lab tests with unirradiated/irradiated samples
Radionuclide leaching rates from UCO kernels	Accelerated tests with failed LEU UCO particles
C-14 content of matrix and graphite	N-14 content in archival matrix and H-451 graphite; C-14 content in irradiated matrix and H-451 graphite
Chemical impurities in H-451 graphite	Chemical analysis of H-451 graphite
Radionuclide leaching rates from graphite	Accelerated lab tests with irradiated H-451 samples

7. FACILITY REQUIREMENTS FOR NGNP TECHNOLOGY PROGRAMS

In order to conduct the R&D programs necessary to satisfy the NGNP DDNs, certain test facilities are needed. Most of the required test facilities are standard equipment for product development and engineering-scale Design Verification & Support (DV&S) activities for power plant- and chemical plant design. Other required test facilities (e.g., materials test reactors, hot cells, etc.) are specialized but established test facilities for nuclear R&D. Finally, some required test facilities (e.g., large-scale coaters, in-pile fission product transport loops, etc.) are highly specialized, almost unique, facilities with their own DV&S requirements.

In general, the NGNP and NHI technology development programs describe (sometimes superficially) the test facilities required to conduct the planned R&D tasks. In some cases, existing test facilities will require extensive refurbishment (e.g., the Irradiated Microsphere Gamma Analyzer (IMGA) at ORNL, etc.). In other cases, new facilities (e.g., an in-pile fission product transport loop, engineering-scale pilot plants for the SI and HTE processes, etc.) must be designed, constructed and commissioned. Most of these refurbishment and new construction activities are based upon established technology and earlier now decommissioned test facilities. Consequently, the GA team expects that such activities will be completed in a timely fashion and represent little risk to the program. However, a few facility design and construction tasks are demanding, effectively first-of-a-kind activities for the NGNP team (e.g., the design and construction of an in-pile fission product transport loop); these activities represent significant cost and schedule risks. Those tasks are identified in this section. Any major test facility deficiencies not identified in the NGNP and NHI TDP are also described in this section.

To summarize, the purpose of this section is to highlight any facility-related, significant cost and schedule risks and any unidentified test facility deficiencies. The purpose is not resolve these issues which is the responsibility of the NGNP Project management. The results of this assessment is summarized in Table 7-1[47] and elaborated in the following subsections.

7.1 Fuel/Fission Products

7.1.1 Fuel Process Development

The AGR fuel development program should include the design, construction, and operation of an integrated, modular, fuel production pilot line capable of demonstrating that high-quality TRISO fuel can be economically mass produced. For the NP-MHTGR program, a conceptual design was completed for an Initial Modular Fuel Process Line (IMFPL) which have served this purpose for the HEU UCO driver fuel for the NP-MHTGR (IMFPL 1992). This conceptual design could serve as the point of departure for the NGNP Project. The IMFPL design for the LEU UCO NGNP fuel would be somewhat simplified because the criticality constraints are less stringent, thereby allowing larger processing equipment, and the security requirements LEU fuel are less severe than for HEU.

[47] This large table is at the end of the section.

7.1.2 Fuel Materials Qualification

A high-temperature (“King”) furnace in a materials test reactor (MTR) is no longer available after the decommissioning of the GA TRIGA facility. Pre-irradiation and post-irradiation fission gas release measurements are the most reliable method for nondestructive measurement of particle failure fraction in fuel compacts. A high-temperature (“King”) furnace should be installed in the TRIGA reactor (Neutron Radiography Reactor, NRR) at INL (Scheffel 2004).

A risk issue that cuts across at least two areas is the lack of hot cells in the Advanced Test Reactor (PPMP 2006).

7.1.3 Radionuclide Transport

7.1.3.1 In-Pile High-Temperature Furnace

An irradiation facility is needed to re-activate irradiated fuel specimens at high temperature prior to postirradiation heating tests to generate an inventory of 8-d I-131. This need is recognized in the AGR Plan/1 (2005). Iodine-131 is the dominant off-site dose contributor during core heatup accidents. There are no data for I-131 release from UCO fuel under core heatup conditions. A high-temperature (“King”) furnace in the TRIGA reactor (NRR) at INL (Section 7.1.2) and used to re-activate irradiated fuel particles and compacts.

7.1.3.2 In-Pile Fission Product Transport Loop

As described in Section 6.1.3.2, the AGR Plan/1 (2005) contains tasks to construct an in-pile fission product transport loop and to perform an in-pile test program. However, the design and construction of the loop are not initiated until FY2013. In-pile loops are extremely complex, engineering-scale test facilities (Hanson 2004b). Such facilities have been successfully designed, constructed, and operated. The COMEDIE loop in the now decommissioned SILOÉ materials test reactor at the CEA research center in Grenoble, France, was the premier example of such a test facility (Blanchard 1986, Medwid 1993).

The technical feasibility of constructing such a facility in the USA (presumably in the ATR) and the attendant costs and schedule must be established far earlier if the design methods for predicting RN transport in the primary circuit are to be validated in a timely manner. The cost and schedule estimates for loop design and construction appear to be extremely optimistic. For example, the AGR/1 (2005) schedule allows one year for loop design and an additional three years for loop construction and commissioning and the conduct of three in-pile tests. As a first approximation, based upon past experience with BD test program in COMEDIE (Medwid 1993), it is estimated that loop design, construction and commissioning would take at least three years and that each test would take an additional year (i.e., at least two years longer than the AGR schedule if given high priority and unconstrained funding).

7.1.3.3 VLPC Simulation Facility

An engineering-scale facility is needed for developing integral test data for validating the design methods used to predict radionuclide transport in the VLPC (Hanson 2007). To provide a realistic radionuclide source, consideration should be given to coupling this facility with the aforementioned in-pile fission product transport loop (Section 7.1.3.2).

7.2 Materials Programs

No test facility deficiencies were identified that would significantly impede the materials R&D programs with the possible exception of US graphite irradiation capacity. The protocols and procedures for ASME and ASTM codification are well established for metals and are currently being embellished for nuclear graphites; this topic is discussed in considerable detail in (NGNP Materials R&D Plan 2005). For HTGR applications, an early experimental challenge for materials qualification was He effects testing. However, technology development at GA, GE and ORNL in the 1970s established experimental techniques for He effects testing in controlled atmospheres. Several laboratory-scale helium loops with appropriate He purification/impurity injection systems will probably be need to be constructed, but this should pose minimal technical risk to the timely production of the required data.

The capacity for graphite irradiation in the USA is practically limited to irradiations in ATR at INL and HFIR at ORNL. In ATR, graphite irradiation capsules have to compete for space with AGR fuel irradiation tests (and other commercial programs as well). Once a graphite TDP is prepared and the test matrices defined, the irradiation capacity requirements can be quantified. A recommendation has already made here to limit the number of candidate graphites to be characterized and to focus on the qualification of an H-451 replacement graphite. If the US capacity for graphite irradiation on a timely schedule is determined to be inadequate, consideration should be given to irradiating leading candidate graphites in HFR Petten and/or in the NIIAR reactors in Russia. The programmatic feasibility of such testing in foreign MTRs should be explored as soon as the graphite TDP has been prepared.

7.3 Energy Transfer Program

The NGNP PPMP (2006) identifies the need for the design and construction of a reasonably large-scale, high-temperature gas test facility for component and materials testing to include the Intermediate Heat Exchanger. Such a facility will also be needed to address of the DV&S DDNs (Section 5.2.3). Currently, there are no operating large-scale He loops in the USA. INL and Brayton Energy, LLC, of Hampton, NH, have prepared a conceptual design of a High Temperature Gas Loop Test Facility to meet this need (HTGL 2006). The HTGL is planned to have a power of 2 MW, a maximum temperature of 950°C, and a maximum pressure of 8.0 MPa. In principle, the design and operation of such a loop should be a low risk endeavor since high-temperature He loops have been successfully operated for more than three decades. Pushing the operating temperature to 950°C is the major design challenge (as it is for the NGNP).

JAEA has an operational He loop. The Helium Engineering Demonstration Loop (HENDEL) was originally constructed to evaluate components for HTTR and to test novel heat exchangers. It will be used to support their SI technology development program. OKBM has three He loops (Table 9-2). CEA and PBMR are currently constructing high temperature, high pressure He loops for component tests.

7.4 Power Conversion System Program

No test facility deficiencies were identified at this writing that would significantly impede the on-going PCU technology demonstration program at OKBM. The design and construction of the requisite test facilities are an integral part of that technology program.

For testing to 950 ^{o}C, several existing test facilities must be upgraded, and two new facilities have to be built; they are listed below. Test specifications will be prepared defining the functional requirements for these facilities.

Facility Upgrades

1. Full-scale test turbocompressor (TC) test facility
2. Lifetime tests on recuperator at higher temperature
3. Hot gas duct tests at higher temperature
4. Material test rigs to operate at higher temperature

New Facilities

1. Turbine blade cooling test facility
2. Facility for testing thermal barrier coatings on turbine blades in high temperature helium.

7.5 Design Verification and Support Program

Additional validation of the nuclear design methods will probably be needed for licensing purposes for the MHR design because of its annular core which uses reflector control rods and its reliance on inherent safety features, especially a strong negative temperature coefficient of reactivity, in contrast to engineered safeguards.

No critical facilities that could be used for these mockup experiments exist in the USA As indicated in Table 7-1, the only currently available facility in the world that could be used is the ASTRA critical at RRC-KI in Moscow, Russia (Kukharkin 2002). The ASTRA facility was used to conduct a series of critical experiments for PBMR. However, these measurements were paid for by ESKOM and the data are not publicly available.

A series of critical experiments are planned for this facility to mockup the annular GT-MHR core that is being designed under a jointly funded DOE/ROSATOM International GT-MHR program for surplus Russian WPu disposition. In addition to a series of room temperature measurements on flux distributions, material reactivity worths, etc., a modification to the ASTRA facility is planned to allow temperature coefficient of reactivity and other nuclear measurements at elevated temperatures (~600 ^{o}C). While plutonium will be used as the fuel, the results would clearly be important for validating the NGNP nuclear design. Repeating these experiments using LEU fuel would also provide a relatively low cost means of obtaining the nuclear validation information required for the NGNP design.

The alternatives to this approach would be the construction of a NGNP critical facility in the USA or modification of the PROTEUS facility in Switzerland. However, a new critical facility would be very expensive, and modifications to PROTEUS would not be cheap.

Although a variety of test rigs will be required for performing the non-nuclear SSC DV&S programs, no major new test facilities are anticipated to be required. Presently, most of the component test rigs are envisioned to be carried in existing national industrial or government test facilities or in international test facilities.

7.6 Hydrogen Production Programs

The requisite, pilot scale- and engineering scale test facilities for both SI and HTE process development will have to be designed and constructed in a timely fashion since such facilities do not currently exist.

7.6.1 Sulfur-Iodine Thermochemical Water-Splitting

Existing facilities at CEA, SNL, and GA are in operation to test and evaluate each of the three sections of the Sulfur-Iodine process. As an example, Figure 7-1 depicts the HI decomposition apparatus. Equipment under construction at General Atomics will integrate the entire process into a continuous, closed loop operation, initially producing hydrogen at rates on the order of 100 standard liters per hour. Non-nuclear heat sources will be used for all testing at GA The GA site will be capable of hydrogen production rates up to a maximum of 1000 standard liters per hour, but it is not suitable for pilot-scale or engineering-scale operations. A new facility must be selected for evaluation of the process at larger scales.

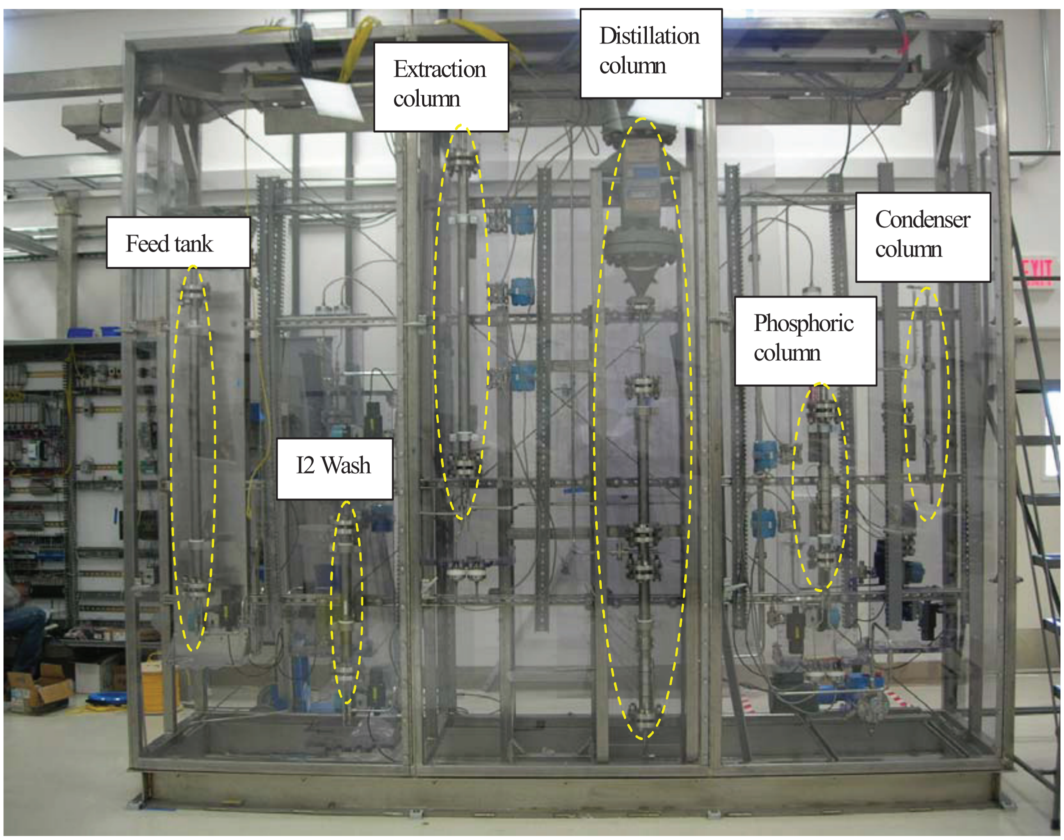

Figure 7-1. GA facility for HI decomposition in the SI process

7.6.2 High Temperature Electrolysis

Toshiba has existing facilities for testing SOE cells and SOE units. Figure 7-2 shows the Toshiba facility for testing SOE units with hydrogen production rates on the order of 100 liters per hour. These facilities will either have to be modified or new facilities constructed to perform testing of multiple units and pilot-scale and engineering-scale modules. A non-nuclear heat source will be used for all testing prior to the NGNP demonstration.

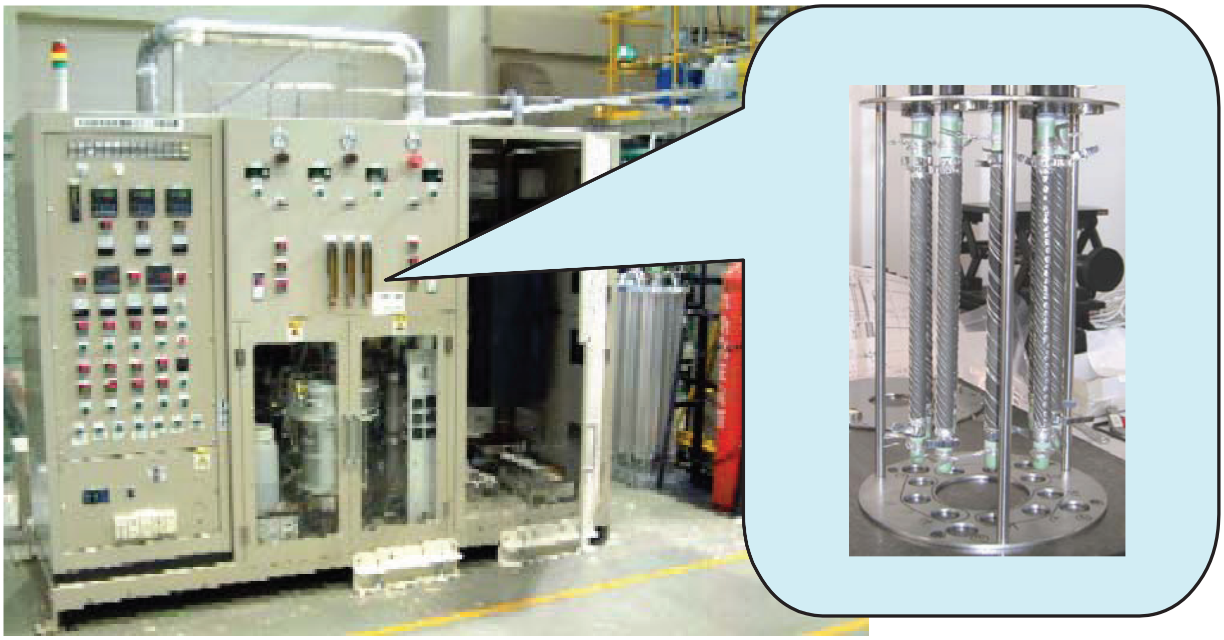

Figure 7-2. Toshiba Facility for Testing SOE Units

7.7 References for Section 7

[AGR Plan/1] "Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program" INL/EXT-05-00465, Rev. 1, Idaho National Laboratory, August 2005.

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[HTGL] "Conceptual Design for a High-Temperature Gas Loop Test Facility," INL/EXT-06-11648, Idaho National Laboratory, August 2006.

[IMFPL] "Conceptual Design Report for New Production-Modular High Temperature Gas-Cooled Reactor Initial Modular Fuel Process Line," IMPL-1B-024, Rev. 1, EG&G, September 1992.

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[NGNP R&D Plan] “Next Generation Nuclear Plant Research and Development Program Plan,” INEEL/EXT-05-02581, Idaho National Laboratory, January 2005.

[NGNP Materials R&D Plan] “Next Generation Nuclear Plant Materials Research and Development Program Plan,” INEEL/EXT-05-00758, Rev. 2, Idaho National Laboratory, September 2005.

[NHI Plan] “Nuclear Hydrogen Initiative Ten Year Program Plan,” US Department of Energy, March 2005.

[PCU TDP] PCU “Technology Demonstration Program Plan, Revision 2005 (Draft),” Product No: 08.03-006.01A, OKBM, 2005 *(Business Confidential).*

Scheffel, W., and J. Saurwein, “GA Recommendation on Replacement Fission Gas R/B Test to Support AGR Fuel Development and Qualification Program,” PC-000518, Rev. 0, General Atomics, May 2004.

Table 7-1. Evaluation of Test Facility Capabilities

DDN Nos.	DDN Category	Test Facility Issue	Programmatic Significance	Recommended Disposition
Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program (AGR Plan/1 2005)				
N.07.01	**Fuel Process Dev.**			
N.07.01.06	Demonstration of mass production of high-quality TRISO fuel.	An integrated modular process line needs to be designed, constructed and operated to demonstrate economical mass production of high-quality TRISO fuel for NGNP.	There is no domestic fuel supplier of TRISO fuel. NFI cannot manufacture 19.8% enriched TRISO UCO fuel which needed for NGNP.	Using the NP-MHTGR IMFPL as a point of departure, design, construct and operate a fuel fabrication facility capable of supplying 19.8% enriched UCO fuel to NGNP.
N.07.02	**Fuel Materials**			
C.07.02.01 - C.07.02.09	Characterization of fuel performance under irradiation and during core heatup accidents.	Hot cell facilities at INL may not be adequate to perform all PIE tasks defined in AGR Plan on the required schedule.	PIE of irradiated fuel specimens essential for characterizing fuel performance and validating performance models.	Upgrade PIE capabilities at INL as required to meet program requirements.
C.07.02.01 - N.07.02.09	Characterization of fuel performance under irradiation and during core heatup accidents.	A high-temperature (“King”) furnace in a materials test reactor (MTR) is no longer available after the decommissioning of the GA TRIGA facility.	Pre-irradiation and post-irradiation fission gas release measurements are the most reliable method for nondestructive measurement of particle failure fraction in fuel compacts.	Install a high-temperature (“King”) furnace in the TRIGA reactor (“NRR”) at INL (Scheffel 2004).
N.07.03	**Radionuclide Transport**			
C.07.03.01 C.07.03.12	Iodine release from fuel compacts during core heatup accidents	An irradiation facility is needed to re-activate irradiated fuel specimens at high temperature prior to postirradiation heating tests to generate an inventory of 8-d I-131. This need is recognized in the AGR Plan/1 (2005).	Iodine-131 is the dominant off-site dose contributor during core heatup accidents. There are no data for I-131 release from UCO fuel under core heatup conditions.	Install a high-temperature (“King”) furnace in the TRIGA reactor (“NRR”) at INL and use it to re-activate irradiated fuel particles and compacts.
C.07.03.14 C.07.03.15 C.07.03.16 C.07.03.17 C.07.03.18	Integral test data to validate design methods for RN transport in reactor core and primary circuit.	An in-pile fission product transport loop needs to be designed, constructed and commissioned (Hanson 2004). AGR Plan/1 (2005) includes construction of loop, but cost and schedule estimate are judged	Design methods for predicting source terms need to be validated.	Establish ASAP technical feasibility of installing an in-pile loop in ATR. Complete conceptual design to provide basis for cost and schedule

DDN Nos.	DDN Category	Test Facility Issue	Programmatic Significance	Recommended Disposition
		unrealistic.		estimate. Evaluate possibility of teaming with RF GT-MHR program to refurbish PG-1 or to construct new loop at NIIAR, Demitrovgrad, RF.
N.07.03.22	Integral test data to validate design methods for predicting RN transport in VLPC	A VLPC simulation facility is needed to generate integral test data to validate the design methods used to predict RN transport in the VLPC during core heatup accidents.	Design methods for predicting source terms need to be validated. If no credit is taken for RN retention in the VLPC, impractical performance requirements will be imposed upon the fuel.	Design, construct and operate a VLPC simulation facility. To provide a realistic fission product source, consider coupling this facility with an in-pile loop.
[Reactor System TDP (TBD)][48]				
C.11.03.12 - C.11.03.21	Core graphite qualification	US capacity for graphite irradiation testing may be inadequate to support NGNP schedule.	A replacement graphite for H-451 must be qualified before a domestic NGNP fuel supply can be established. It is highly desirable that the graphite for the initial core be manufactured in the USA.	Prepare a graphite TDP and focus on qualifying a replacement graphite for H-451. If necessary, determine programmatic feasibility of graphite irradiation testing in foreign MTRs.
C.11.03.51	Integral Nuclear Data Measurement at Temperature for GT-MHR Physics Methods Validation	A mockup MHR annular core critical facility, capable of reaching several hundred degrees centigrade is required for these measurements. No such facility exists in the USA. The ASTRA critical in RRC-KI appears to be the only available option.	Greater uncertainties applied to control rod worth and temperature feedback effects on reactivity. This could impact plant safety and require design changes	Conduct critical experiments in the ASTRA facility with LEU fuel. Arrange for access to results from planned WPu fueled GT-MHR criticals
C.11.03.52	Critical Experimental Data for GT-MHR Physics Methods Validation	Same test facility issue as for DDN C.11.03.51. In particular control rod worth measurements in an annular core mockup are needed.	Same impact as for DDN C.11.03.51	Same as for DDN C.11.03.51 above
[Energy Transfer Technology Development Plan (TBD)][49]				
N.13.01	**[PHTS Circulator]**			

[48] No TDP for the Reactor System has been prepared to date for the NGNP Project.
[49] A Energy Transfer TDP is to be prepared (Appendix E, PPMP 2006).

DDN Nos.	DDN Category	Test Facility Issue	Programmatic Significance	Recommended Disposition
	DV&S testing for 950 ^{o}C operation and equipment qualification	A large-scale, high-temperature gas test facility is needed for component and materials testing.	DV&S tests will be needed to verify FOAK design of high temperature circulator.	Complete final design of high temperature gas loop in USA. Evaluate programmatic options for testing in foreign high temperature He loop.
N.13.02	**<u>Intermediate Heat Exchanger (IHX)</u>**			
	DV&S testing for 950 ^{o}C operation and equipment qualification	A large-scale, high-temperature gas test facility is needed for component and materials testing.	DV&S tests will be needed to verify FOAK design of IHX which is high risk and mission critical.	Complete final design of high temperature gas loop in USA. Evaluate programmatic options for testing in foreign high temperature He loop.
PCU TDP (2005)				
N.41.00	**PCS**			
	Data for 950 ^{o}C inlet PCS operation	Existing test facilities at OKBM must be modified, and some new test facilities need to be built.	Qualify PCS for operation at 950 ^{o}C.	Upgrade existing test facilities and build new facilities at OKBM as required to qualify PCS for 950 ^{o}C inlet operation.
[Energy Transfer Technology Development Plan (TBD)]				
N.42.02	**<u>Isolation Valves</u>**			
	DV&S testing for 950 ^{o}C operation and equipment qualification	A large-scale, high-temperature gas test facility is needed for component and materials testing.	DV&S tests will be needed to verify FOAK design of high temperature isolation valve.	Complete final design of high temperature gas loop in USA. Evaluate programmatic options for testing in foreign high temperature He loop.
NHI Plan (2005) – SI Process				
N.44.01	**<u>[Sulfuric Acid Decomposition]</u>**			
N.44.01.01	Catalyst Performance	Lab-scale work can be done in current facilities. Appropriate test facilities need to be identified for pilot-scale and engineering-scale testing	Better understanding of actual plant capacity factors and capital/operating costs	Construct or select new facility as process is scaled up from laboratory

DDN Nos.	DDN Category	Test Facility Issue	Programmatic Significance	Recommended Disposition
N.44.02	**[Bunsen Reaction]**			
N.44.02.01 - N.44.02.02	Reaction Physical Chemistry Data	Lab-scale work can be done in current facilities. Appropriate test facilities need to be identified for pilot-scale and engineering-scale testing	Optimized reactor design, reduced cost, reduced side reaction	Construct or select new facility as process is scaled up from laboratory
N.44.03	**[Hydrogen Iodide Decomposition]**			
N.44.03.01	HI/H_2 Membrane Separation	Lab-scale work can be done in current facilities. Appropriate test facilities need to be identified for pilot-scale and engineering-scale testing	Reduced cost compared to refrigerated phase separation	Construct or select new facility as process is scaled up from laboratory
N.44.03.02	Refined Thermodynamic Model	Lab-scale work can be done in current facilities. Appropriate test facilities need to be identified for pilot-scale and engineering-scale testing	Improved design reliability	Construct or select new facility as process is scaled up from laboratory
N.44.04	**[Materials Compatibility]**			
N.44.04.02	Equipment Manufacturability	Facility is needed for testing.	Optimal design, reduced cost	Construct or select new facility.
NHI Plan (2005) – HTE Process				
N.45.01	**SOE Cells**			
N.45.01.01 N.45.01.02	SOE Cell data	None.	Required for HTE plant design.	Use existing facilities.
N.45.02	**SOE Units**			
N.45.02.01 N.45.02.02	SOE Unit data	Facility is needed for multiple-unit testing.	Required for HTE plant design.	Use (or modify) existing facilities.
N.45.03	**SOE Modules**			
N.45.03.01 N.45.03.02 N.45.03.03	SOE Module data	Appropriate test facilities need to be identified for pilot-scale and engineering-scale testing. Facilities will produce significant quantities of hydrogen and require significant quantities of heat.	Required for HTE plant design.	Modify existing facilities or construct new facilities.

DDN Nos.	DDN Category	Test Facility Issue	Programmatic Significance	Recommended Disposition
N.45.04	**HTE Plant Supporting Equipment**			
N.45.04.01 N.45.04.02	SOE heat exchanger design and performance data	Appropriate test facilities need to be identified for testing heat exchanger components.	Reliable steam generator/superheater and recuperators are needed for HTE plant design.	Develop more detailed designs and determine if vendors can supply required design and performance data.

8. QUALITY ASSURANCE PROGRAM

The quality assurance (QA) requirements for the NGNP Project, as related to the R&D programs, are defined in the PPMP (2006) as follows:

> "The Quality Assurance requirements for specific-tasks, such as the NGNP R&D activities, should be specified in their project-specific QAPs and project-specific Technical Specifications. The R&D project specific QAPs will include the management controls commensurate with the project work scope and importance to the NGNP Project goals and objectives.
>
> "In the document hierarchy, the NGNP QAP is the top-level Quality document. The NGNP QAP is subtier to the NGNP PPMP, and NGNP R&D project QAPs are subtier to the NGNP QAP. Partnering laboratories for R&D and subcontractors will be under the requirements of the respective INL R&D project QAP or will develop their own QAP that is subtier to the respective INL QAP.
>
> "When the NGNP R&D phase begins, the development of design specifications to be used in the design phases, including Conceptual Design, or construction, the NGNP Quality Assurance Requirements document will require full compliance to ASME NQA-1-1997, Part 1 and Subpart 2.7, as well as the ASME Construction Code Section III, "Nuclear Power." This will also involve the preparation and submittal of a license application request to the NRC to begin the construction phase activities and the application of 10 CFR 50, Appendix B, "Quality Assurance," which is the NRC licensing quality assurance requirements for both a Part 50 or Part 52 license. At the time of NRC license application, the NGNP Project should decide to either continue to use the 1997 version of NQA-1 or chose to use the latest issued edition of the NQA-1 Standard, which is currently 2003."

In principle, the GA team strongly endorses this overall QA approach for the NGNP technology programs. While QA is always an important consideration in performing R&D, past experience has demonstrated - sometimes painfully - that rigorous application of appropriate QA protocols is essential for successful completion of R&D that will be used for the design and licensing of nuclear facilities, including both nuclear power plants and nuclear fuel facilities.

In addition to the QA requirements for NGNP technology programs, there are two other critically important QA issues that will need to be addressed. First, there is an extensive of amount of existing international data that could be used to partially satisfy the NGNP DDNs identified herein. Some of these data, while judged to be of high technical quality and reliability, do not have a formal QA pedigree which meets all of the requirements of NQA-1. Most notable in this category is the German database for fabrication and performance testing of high-quality LEU UO_2 TRISO fuel. From extensive direct interaction with the German researchers under the past US/FRG Umbrella Agreement, GA is confident that these German data are highly reliable; however, the earlier German QA protocols were different from NQA-1 (stated simplistically, the German approach was to certify people and not procedures). The approach taken on previous US HTGR programs, including the commercial GT-MHR program, was to assume that these German data could be used for model development, etc., but that at least some independent, fully NQA-1 compliant, data would need to be generated by the USA, especially for code validation. This same approach is recommended for the NGNP Project.

The second closely related issue is the formal QA pedigree of any future data produced by foreign participants in the NGNP Project that would be used for design and licensing. In this regard, the Russian GT-MHR program is committed to meeting the QA requirements of ISO 9000/9001 (OKBM is already ISO 9000 certified), and the equivalence between ISO 9000/9001 and ASME NQA-1 has been established Subpart 4.3, Guide to Modification of an ISO 9001-2000 Quality Program to Meet NQA-1-2000 Requirements, Tables 200-1 through 200-18 (Part IV: Nonmandatory Appendices – Positions and Applications Matrices, ASME NQA-1-2004, Quality Assurance Requirements for Nuclear Facility Applications). A review of the cross-reference of the sections in each of the standards reveals some NAQ-1-2004 sections have no corresponding section in ISO 9001-2000. In addition, NQA-1-2004 sections are very specific in describing the requirements compared to the generalizations in ISO 9001-2000.

One solution to the dilemma associated with using these foreign sources may be to use Appendix 3.1, of ASME NQA-1-2004 to qualify the data. Appendix 3.1 provides nonmandatory guidance on the qualification of existing data to be used in support of achieving safe, reliable, and efficient utilization of nuclear energy, and the management and processing of radioactive materials. It includes selecting the Data Sets for Qualification, the Data Qualification Process, Qualification Methods, and Documentation of Results (Part III: Nonmandatory Appendices, ASME NQA-1-2004, Quality Assurance Requirements for Nuclear Facility Applications).

9. POTENTIAL FOR INTERNATIONAL COLLABORATION

There is an impressive history of successful international collaboration on HTGR development, especially in the fuel, fission products and graphite areas. Arguably, the first major international cooperation on HTGR development – the Dragon Project – remains the most ambitious and successful one (ironically, the US government chose not to formally participate in Dragon). The OECD-sponsored, the Dragon Project did an extensive amount of pioneering R&D, especially in the areas of fuel, fission products, and coolant chemistry, in addition to designing, constructing and operating the first HTGR (Ashworth 1978).

One obvious and important difference in the on-going international Modular HTGR programs is the choice of core design with the RF GT-MHR program and the Japanese program having selected a prismatic core and the PBMR and Chinese programs having chosen a pebble-bed core. (The core design for NGNP has not been officially selected at this writing; the GA team has chosen a prismatic design, consistent with all previous US MHR designs.)

At first consideration, this basic core design difference might seem to be a major impediment to international collaboration. While this difference is indeed a complication in some regards, history is again reassuring and encouraging. The USA and Germany had a very productive cooperative program for gas-cooled reactor development, beginning in the late 1970s and continuing until the FRG HTR program was terminated in the late 1980s because of unfavorable nuclear politics, exacerbated by the Chernobyl accident. The fact that the US designs employed prismatic cores and the German designs (usually) employed pebble-bed cores resulted more in a friendly rivalry than in insurmountable obstacles to collaboration. In fact, the US/FRG cooperation in the fuel, fission products and graphite area (e.g., GA-C18856, 1987) – seemingly areas that would highlight core design differences – was the most extensive and the most successful.

9.1 Russian GT-MHR Program

As introduced in Section 2.2.5, the International GT-MHR is being developed under a joint USDOE-NNSA/ROSATOM program for the purpose of destroying surplus RF weapons plutonium. The reference plant design is very similar to the GA commercial GT-MHR design with an improved PCU design. A preliminary design has been completed, and construction of a bench-scale facility for fabrication of TRISO-coated, PuO_{2-x} test fuel is nearly complete. The lead design organization, OKBM, is a member of the GA team, and the OKBM PCU is part of the GA “reference” preconceptual design. The goal is to have a demonstration plant, consisting of two 600 MW(t) modules, in operation by 2019.

9.1.1 Technology Demonstration Program Plan

While the primary long-term goal is to design, develop and demonstrate a GT-MHR for Pu disposition, projected funding limitations caused a restructuring of the program to focus on demonstrating high-risk technologies, namely, the TRISO fuel and turbocompressor, along with the development of selected long-lead component technologies. The following objectives are described in the Technology Demonstration Program Plan (TDPP 2005):

Pu TRISO Fuel

- Demonstrate Pu TRISO fuel can be made in Russia to the required quality standards, can meet its performance and fission product retention requirements, and will achieve the burnup level that satisfies Pu-239 conversion and nonproliferation requirements of the GT-MHR design.

Power Conversion Unit

- Identify and resolve key technical issues that represent PCU-related barriers to the deployment of the first GT-MHR;
- Develop and demonstrate key subsystems and components of the GT-MHR PCU;
- Confirm the feasibility and performance of the PCU through a full-scale integrated test of the turbocompressor;
- Accomplish the above with optimal management of development cost and risk.

The scope of the PCU Development Program (PCU TDP 2005) includes the design, development and verification testing of PCU systems and components from the present through the decision to build the first GT-MHR plant.

Vessel System

- Demonstrate vessel material performance and manufacturability.

Reactor System

- Demonstrate core graphite can be manufactured in Russia to the GT-MHR specifications;
- Validate Reactor Physics and Fission Product transport models and codes.

The total cost of this technology demonstration program, for performance of the work in Russia, is estimated at $149M (2005 US$). Cost of work in Russia is assumed to be funded on the basis of 50% cost share between DOE/NNSA and ROSATOM. The technology demonstration program is structured with the assumption that a combined USA and RF funding level of $10-20M/per year will be provided.

The RF GT-MHR and the NGNP share many common DDNs, and much of the on-going RF technology program would be directly supportive of the NGNP Project. The OKBM PCU design is part of the GA preconceptual design. As discussed in Section 5.2.4, the OKBM design will have to be modified for 950 ^{o}C operation (presumably by the addition of blade cooling) and to address the issues raised by the Rolls-Royce independent review (PCDSR 2007). Much of the fuel, fission product and graphite technology programs should be directly relevant as well.

In addition to having common DDNs that the RF TDPP (2005) could address for the NGNP Project, DOE/NNSA is providing half of the funding for the RF GT-MHR program. As a result, many of the intellectual property issues and QA pedigree issues that typically complicate international collaboration on nuclear construction projects should, in principle, be more tractable. Consequently, it is recommended that, as a follow-on task to the present preconceptual design studies, an in-depth evaluation be made of the feasibility and benefits of utilizing RF R&D programs and test facilities to satisfy a number of NGNP DDNs.

9.1.2 Russian Test Facilities

There are a number of Russian test facilities that could be used to perform tests necessary to satisfy NGNP DDNs. The RF facilities that might be used to compensate for some of the facility deficiencies identified in Section 7 are described briefly below.

9.1.2.1 Fuel/Fission Products

Test facilities for the irradiation of coated-particle fuel are being established at the "Scientific Research Institute for Atomic Reactors" (NIIAR), Dimitrovgrad, RF, as part of the DOE/ROSATOM GT-MHR program (RF Fuel Plan 2005). The use of two NIIAR reactors is planned: the **SM-3** reactor and the **RBT-6** reactor. These reactors provide a variety of test channels and operating environments; the testing capabilities of these two reactors are summarized in Table 9-1.

Table 9-1. Irradiation Testing Capabilities of RF NIIAR Reactors

Parameter	Required for GT-MHR	Value			
		SM-3*			RBT-6
		Type 1 Channels	Type 2 Channels	Type 3 Channels	
Core height, mm		350	350	360	350
Diameter of channels, mm		64	64	64	64
Number of channels		4	4	4	8
Number of ampoule/channel		3	3	3	1/3
Number of compacts/ ampoule		4	4	4	12/(300 CP)
Fuel compact power, kW - average - maximal	 0.2 0.6	0.6-0.45	0.3-0.15	0.6-0.45	0.34-0.15
Max. full neutron flux, x10^{14} n/cm^{2}s	2.8	~15	3.1	5.4	2.0
Max. fast (E>0.18 MeV) neutron flux, x10^{14} n/cm^{2}s	0.6	3.1	0.65	0.62	0.53
Time needed for full burnup (EFPD/calendar)	750/	300/500	710/1180	360/600	800/1140
Time needed for full fast fluence (EFPD/calendar)	750/	150/250	710/1180	710/1180	900/1290

* Type 1 Channels: 1st reflector row with 1-2 mm Cd screen; Type 2 Channels: 2nd reflector row with 4 mm Hf screen; Type 3 Channels; 2nd reflector row with 2 mm Hf screen.

The SM-3 reactor has higher neutron flux locations and can be used for testing of statistically significant numbers of particles in compacts and to produce irradiated compacts for accident testing. The significantly lower flux RBT-6 can be used to test fuel compacts and loose particle samples and fuel material samples to obtain fuel material irradiation characteristics, fission product transport data, and produce irradiated material for special tests. For GT-MHR Pu fuel, full burnup (~70% FIMA) and full fast fluence (4×10^{25} n/m^2) can be reached in a relatively short time (~1 yr) in the inner positions of SM-3; however, it is considered prudent to limit the irradiation acceleration to less than a factor of three higher than the GT-MHR. This limitation implies that full exposure can be accomplished in about 300 to 750 calendar days in the SM-3.

Coated-particle fuel irradiation capsules can be inserted into test "channels" in these reactors. Each test train is made up of "ampoules" (cells). Four channels in SM-3 are suitable for irradiation testing of coated particles. The irradiation test train currently being designed for the GT-MHR program consists of three ampoules. Each of the ampoules can accommodate four compacts; consequently, a maximum of 12 compacts can be tested in each channel and a maximum of 48 compacts can be tested simultaneously in the four SM-3 channels. Ampoules are also currently being designed for the RBT-6 reactor. These new facilities will permit multi-cell irradiations of loose particles and compacts; design details are not available at this writing. To reach full burnup and fast fluence simultaneously, it is necessary to reduce the thermal flux by using neutron shields of materials such as hafnium.

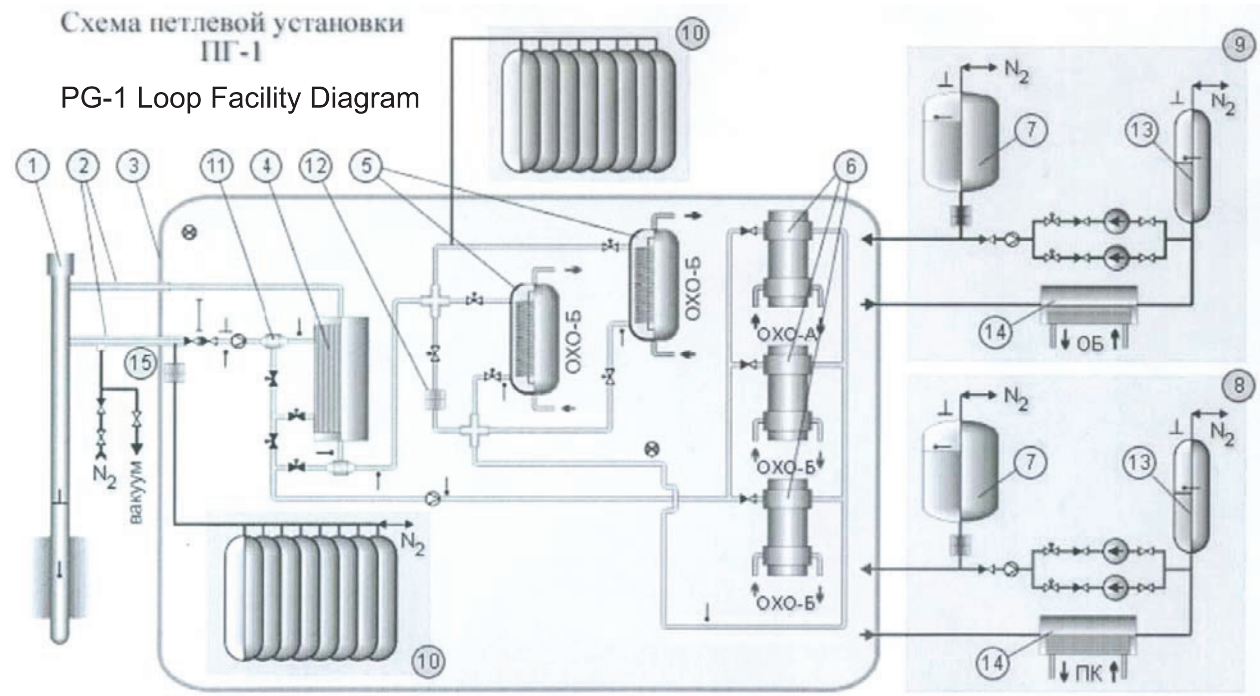
NIIAR has extensive hot cell facilities and years of experience of performing postirradiation examination of irradiated LWR, RBMK, and fast reactor fuel;[50] however, they have not previously irradiated or performed PIEs of coated-particle fuel. Consequently, the specialized PIE equipment needed for the PIE of TRISO fuel is currently being constructed. Postirradiation heating furnaces will also be constructed. In addition to fuel irradiations, RF graphites will also be irradiated at NIIAR, and it may be possible to include candidate NGNP graphites in the test matrix.

Analogous to the AGR program, the RF TDPP (2005) calls for a series of in-pile loop tests to validate the design methods for predicting radionuclide source terms. No operational, in-pile fission product transport loop currently exists in the Russian Federation which is capable of performing the required integral tests. The PG-100 loop at RRC KI, which would have been capable of meeting many of the performance requirements, has been permanently decommissioned. The PG-1 loop at NIIAR is currently not operational but, in principle, could be recommissioned.

The PG-1 loop, located in the MIR reactor at NIIAR, was designed to be a high-temperature, high-pressure, high-flow gas loop to investigate fission product transport; however, it was designed to operate with N_2O_4 as the coolant rather than helium (Osipov 2000). A schematic of the loop is shown in Figure 9-1. Significant redesign and refurbishment would be required to convert the PG-1 to a helium loop. It cannot be determined definitively at this writing whether or not the PG-1 loop could be sufficiently redesigned and refurbished to meet the performance requirements needed for the NGNP program. It appears capable of operating at or near the required service conditions; its greatest limitation currently appears to be physical constraints (i.e., a small space envelope for the test components). A comprehensive review of the PG-1

[50] http://www.niiar.ru/eng/docl.htm

loop design and its potential to support the NGNP program needs to be performed expeditiously.



1 - Loop channel; 2 - Secure shrouds; 3 - Protective shell; 4 - Recuperator; 5 - Coolers; 6 - Gas circulators; 7 - Hydraulic accumulators of emergency cooling systems; 8 - Equipment cooling system (OXO-A); 9 - Equipment cooling system (OXO-B); 10 - Channel emergency cooling system; 11 - Mixer; 12 - Throttle; 13 - Pressurizers; 14 - Heat exchangers.

Figure 9-1. Schematic of PG-1 Loop at NIIAR

The alternative is to construct a new in-pile loop in the MIR reactor which would utilize as much of the PG-1 hardware and auxiliary systems (e.g., gas supply and storage, instrumentation, etc.) as practical.

9.1.2.2 Power Conversion System

The extensive test facilities at OKBM for conducting the PCU TDP (2005) are illustrated in a viewgraph presentation included here as Appendix B.

9.1.2.3 DV&S Test Facilities

The ASTRA critical facility, shown in Figure 9-2, at the Kurchatov Institute in Moscow is designed for the experimental investigation of the nuclear characteristics of HTGR reactors. Presently, ASTRA is configured in the form of a cylinder with an internal cavity to form a core. The central cavity is filled with spherical elements since critical experiments have been performed for PBMR under contract with ESKOM (Kukharkin 2002). In these experiments the central reflector was originally composed of unfueled graphite pebbles (“dynamic reflector”) but now has been replaced a solid graphite central reflector. These cold critical experiments were the first performed with an annular core geometry. Most of the PBMR test results have not yet been published in the open literature.

The RF GT-MHR program is planning to perform critical experiments in ASTRA with an annular core geometry representative of the GT-MHR (TDPP 2005). The first tests will be done with the existing pebble fuel (10%-enriched UO_2) and will measure the worths of reflector control rods. Measurements will be made over a temperature range of 20 to 600 ^{o}C. Like the cold criticals before them, these tests will be first for an annular core geometry at elevated temperatures. Future tests are planned with Pu fuel when it becomes available in sufficient quantities.

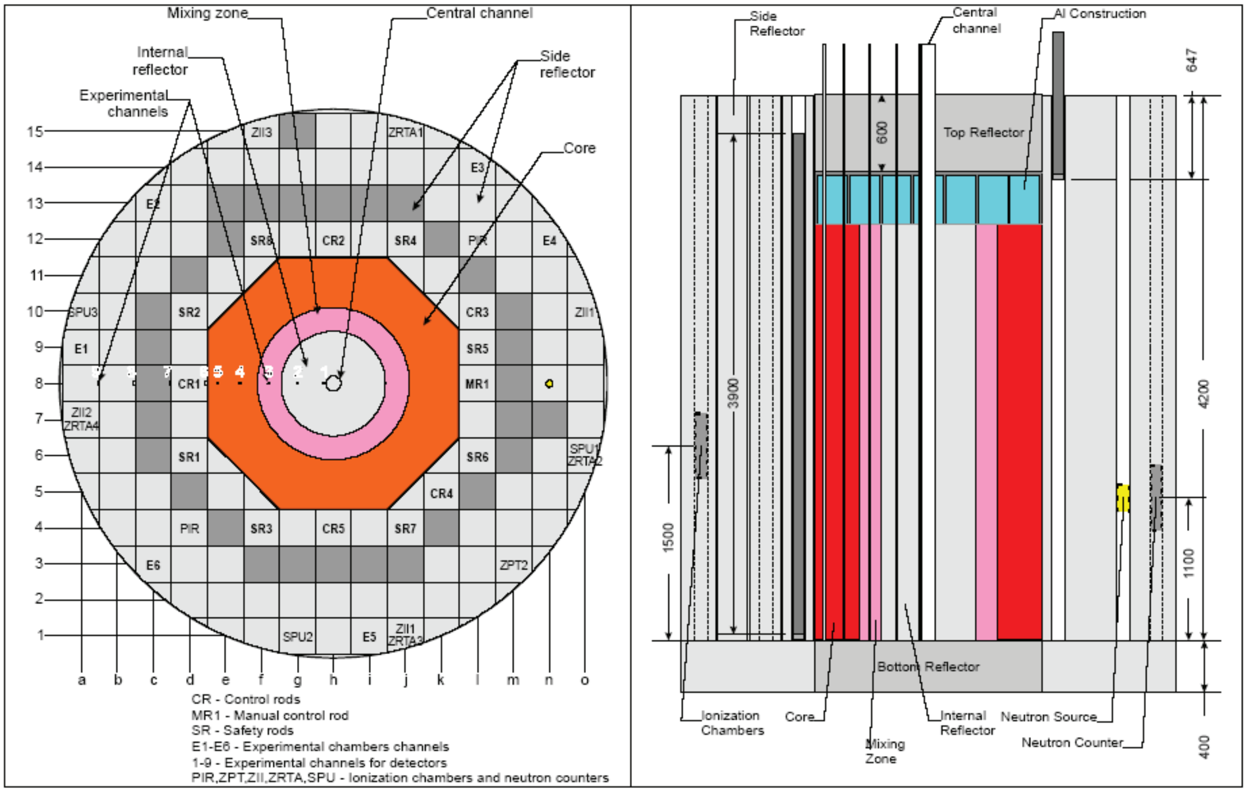


Figure 9-2. ASTRA Critical Test Facility

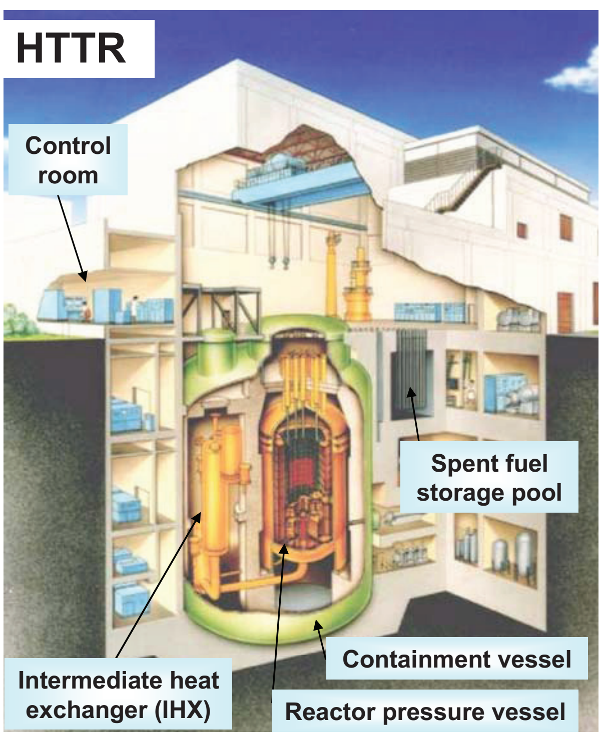
OKBM has three high temperature, high pressure helium loops suitable for performing a number of tests required to satisfy DV&S DDNs. The operating characteristics for these loops are summarized in Table 9-2. The largest loop (ST1312) is capable of accommodating large components at 950 ^{o}C and appears particularly suitable for testing key NGNP components, including candidate IHX and recuperator designs.

Table 9-2. OKBM He Test Loops

Loop	Working Fluid	Peak Temperature (^{o}C)	Pressure (MPa)	Power (MW)	Main Application
ST1565	He	950	5.0	0.5	General, HPS
ST1312	He	950	5.0	12-15	IHX, steam generator
ST1383	He	350	5.0	6.0	He circulator

9.2 Japanese MHR Program

Japan has had an active interest in HTGR technology for decades. Presently, VHTR and nuclear hydrogen design and technology development is being conducted by the Nuclear Applied Heat Technology Division of the Japan Atomic Energy Agency. JAEA also operates the 30-MW(t) High Temperature Engineering Test Reactor (HTTR), located at the JAEA Oarai Research Establishment. The HTTR is a helium-cooled, graphite-moderated HTGR that uses prismatic graphite fuel elements. Full-power operation with a coolant outlet temperature of 850 °C was achieved in December 2001. Operation with a coolant outlet temperature of 950 °C was accomplished in 2004 and extended (50 day) operation with 950 °C outlet temperature was performed in 2007. The primary purposes for the HTTR are: (1) to establish basic VHTR technology and demonstrate VHTR inherent safety, (2) demonstrate utilization of nuclear heat to produce hydrogen, and (3) to irradiate VHTR fuel and materials under prototypical conditions. Figure 9-3 shows the cut-away views of the HTTR reactor building and pressure vessel/reactor core. The pressure vessel is 13.2 m in height and 5.5 m in diameter, and was manufactured from 2¼Cr-1Mo steel. Major design parameters for the HTTR are given in Table 9-3.



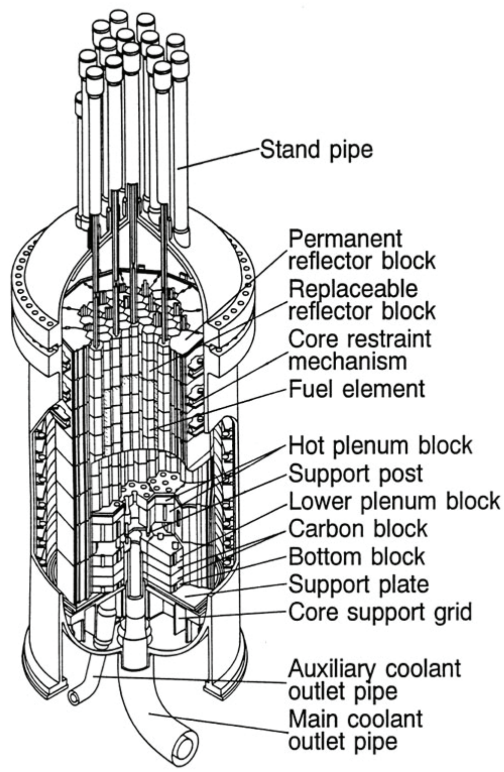


Figure 9-3. JAEA High Temperature Engineering Test Reactor

Table 9-3. HTTR Major Design Parameters

Thermal power	30 MW
Core outlet temperature	850/950 °C
Core inlet temperature	395 °C
Fuel	Low enriched UO_2 (6wt%)
Fuel element type	Prismatic block
Fuel loading	Off-load, 1 batch
Core diameter	2.3 m
Core height	2.9 m
Average core power density	2.5 MW/m^3
Core flow direction	Downward flow
Coolant flow rate	10.2 kg/s (950°C operation)
Primary coolant pressure	4.0 MPa

The JAEA Applied Heat Technology Division includes the following groups:

- HTGR Cogeneration Design and Assessment Group
- High Temperature Fuel and Materials Group
- HTGR Performance and Safety Demonstration Group
- Iodine-Sulfur (IS) Process Technology Group
- HTGR-IS Coupling Technology Group

The JAEA plan for developing nuclear hydrogen production technology is shown in Table 9-4. JAEA eventually plans to couple a hydrogen production plant to the HTTR using 10 MW of heat supplied from the HTTR IHX. The IHX has already been installed in the HTTR and is a helical coil design manufactured from Hastelloy-XR (see Figure 9-4). Figure 9-5 shows various JAEA facilities that support IS process development.

Table 9-4. JAEA Nuclear Hydrogen Development Plan

	Bench-scaled Test	**Pilot Test**	HTTR Test nuclear demonstration
Hydrogen production rate	~ 0.05 m^3/h	~ 30 m^3/h	~ 1000 m^3/h
Heat supply	Electrical heater	Heat exchanger with helium gas (Electrical heater 0.4MW)	Heat exchanger with helium gas (Nuclear heat 10MW)
Material of chemical reactors	Glass	Industrial material (SiC, coated)	Industrial material
Pressure of chemical process	Atmospheric pressure	High pressure (up to 2MPa)	High pressure (up to 2MPa)
Time	FY 1999 - 2004	FY **2007** – 2012 (under planning)	FY 2011 – 2016 (under planning)

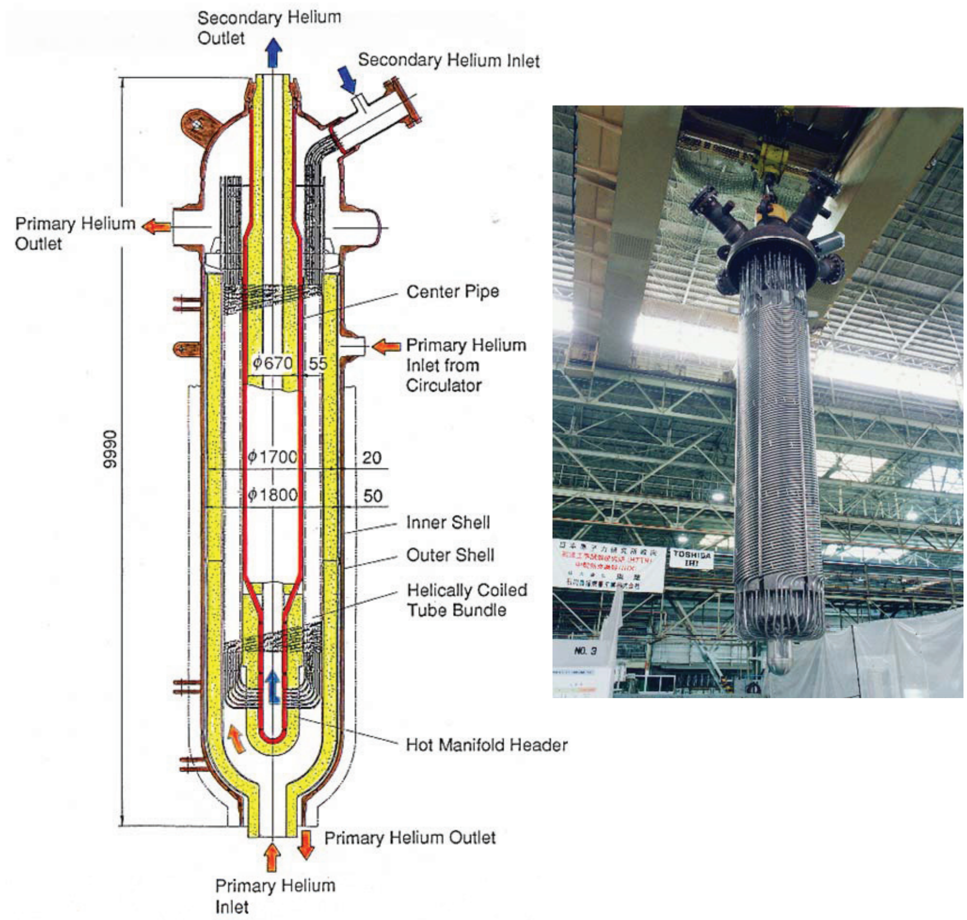


Figure 9-4. HTTR Intermediate Heat Exchanger

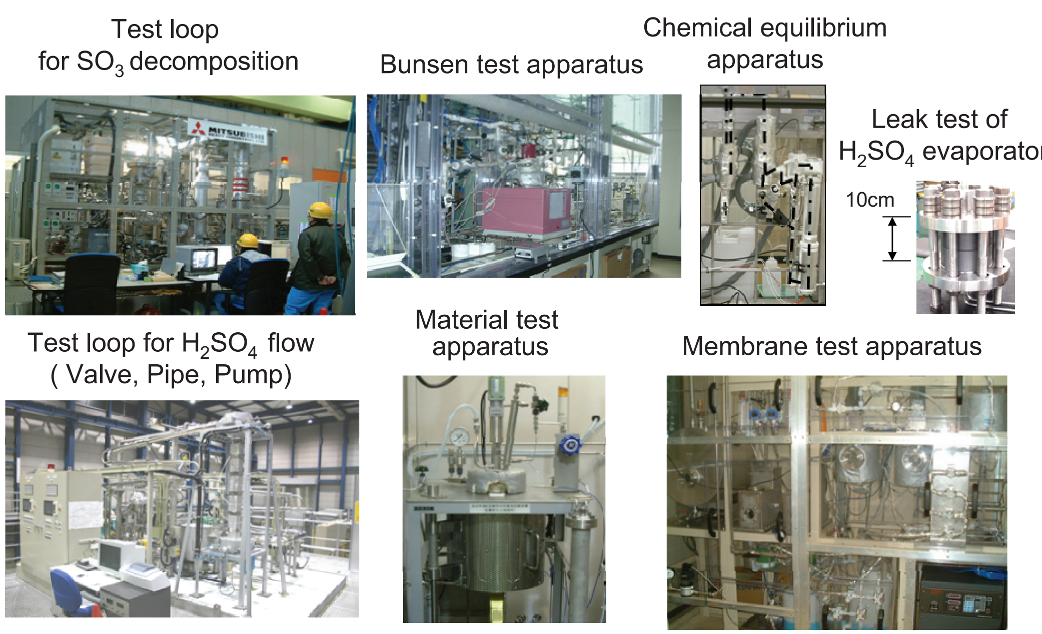


Figure 9-5. JAEA Facilities for IS Process Development

Nuclear Fuel Industries (NFI) of Japan manufactured the SiC-TRISO fuel for the HTTR. JAEA is currently developing ZrC as a coating material to replace the SiC layer of the TRISO coating. ZrC can operate at higher temperatures than SiC, and it may be a more effective barrier to diffusive release of Ag-110m at high temperatures. Figure 9-6 shows the JAEA ZrC coating development facility.

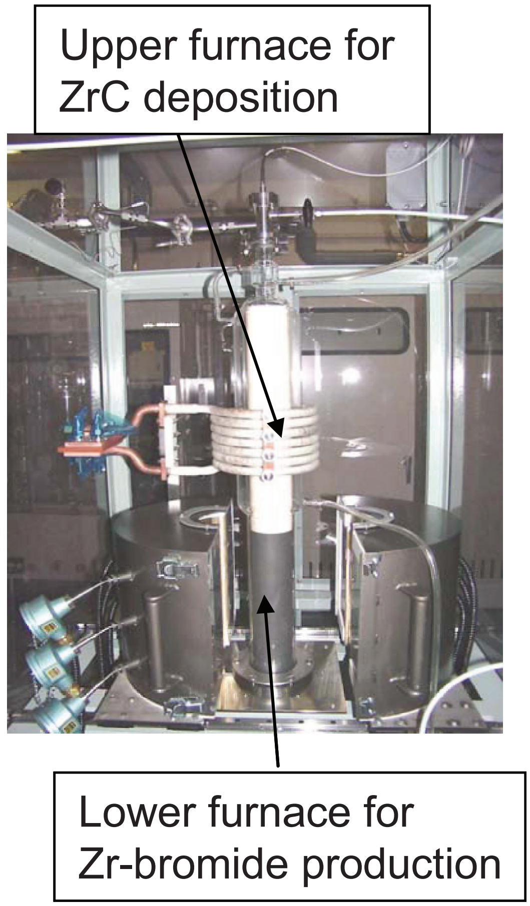


Figure 9-6. JAEA Facility for ZrC Coating Development

JAEA has developed commercial HTGR concepts for production of electricity and cogeneration of electricity and hydrogen. Figure 9-7 shows the JAEA GTHTR300C cogeneration plant. The reactor operates at 600 MW(t), and the design is similar to that of the GT-MHR. JAEA has performed compressor and magnetic bearing tests to support the JAEA GTHTR300C design. Figure 9-8 shows the 1/3-scale model that was used to generate design data for the helium gas compressor.

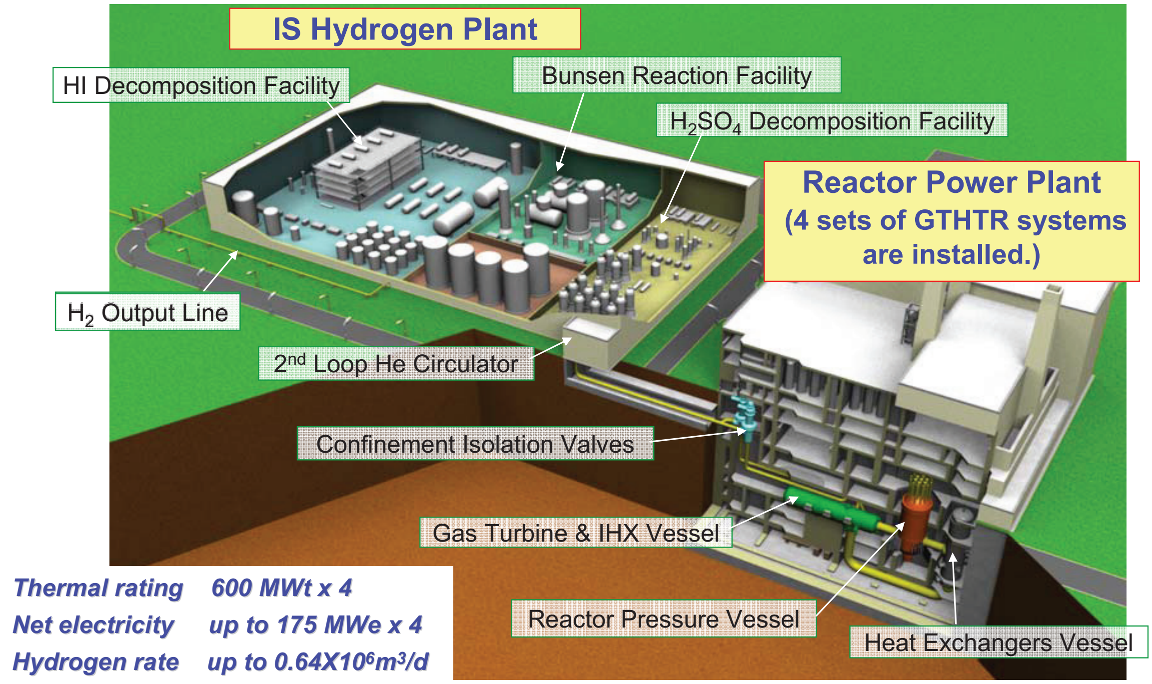


Figure 9-7. JAEA GTHTR300C Cogeneration Plant

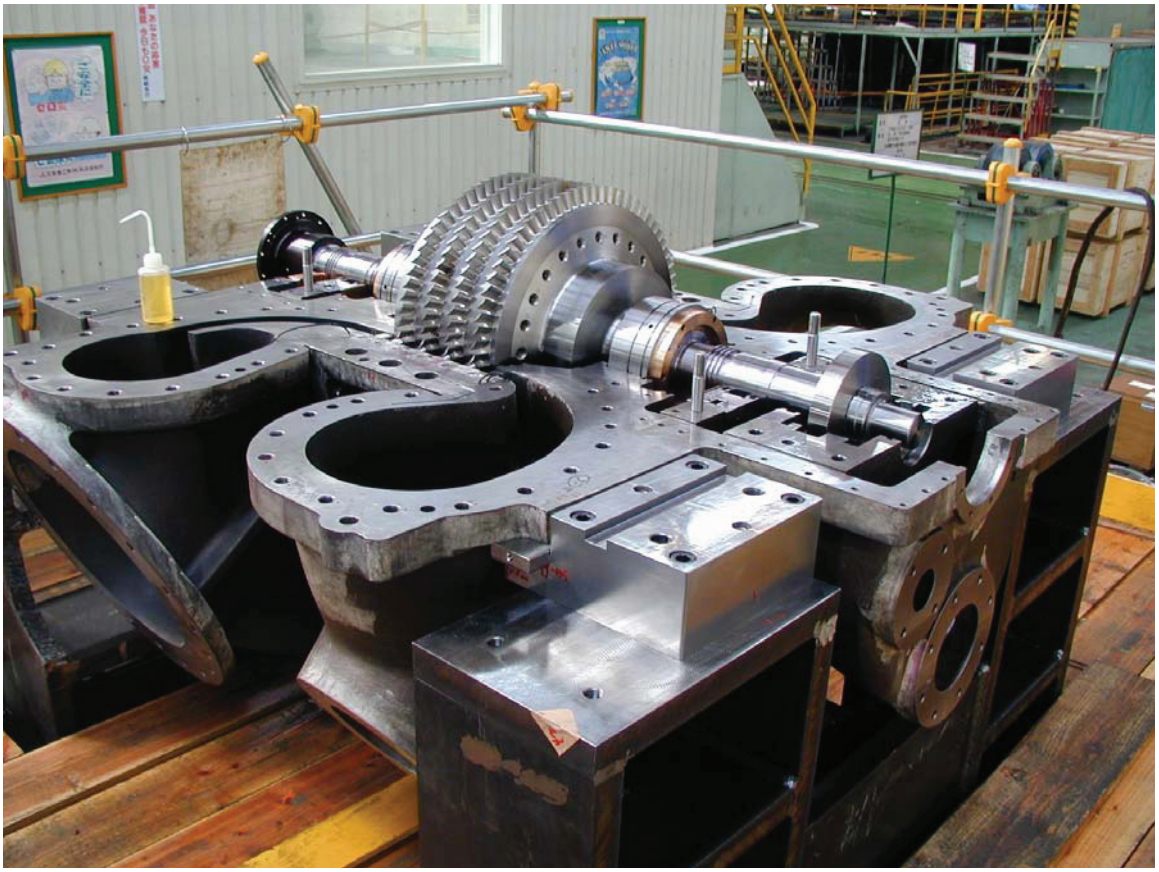


Figure 9-8. Helium Gas Compressor 1/3-Scale Model

Because Japan has very limited nuclear fuel resources, the current emphasis in Japan is to develop the sodium-cooled, fast breeder reactor, and the budget in Japan to support HTGR design and technology development has been reduced in recent years. However, the NGNP project provides an excellent opportunity for collaboration with Japan and to benefit from the Japanese experience in HTGR design and technology. The NGNP project could also benefit from using existing Japanese facilities (e.g., using the HTTR to provide data on tritium/fission product transport and the NFI fuel manufacturing facilities to provide the NGNP initial core).

9.3 Korean MHR Program

In 2004, the Republic of Korea (ROK) initiated a project to develop nuclear hydrogen production using the VHTR and the SI process. VHTR design and technology development is being performed by the Korea Atomic Energy Research Institute (KAERI), and development of the SI process is being performed by the Korea Institute of Energy Research (KIER) and the Korea Institute of Science and Technology (KIST). DOOSAN Heavy Industries & Construction is also participating in the project, which is known as the Nuclear Hydrogen Production and Technology Development and Demonstration Project (NHDD).

KAERI is the lead organization for the NHDD project and has solicited international participation, including establishing a partnership with General Atomics. In August 2005, a Memorandum of Understanding was signed between GA and KAERI/DOOSAN, which included establishing Nuclear Hydrogen Joint Development Centers (NHJDC) in both San Diego and Daejeon, Korea. Current areas of collaboration include SI process development and modelling, VHTR core design and optimization, vessel cooling, fuel performance and fission product transport, tritium source terms and impacts on hydrogen production, fuel manufacturing, availability/reliability, seismic analyses, availability/reliability assessments, and investigation of deep-burn fuel cycles. KAERI is also an active participant in Generation IV VHTR activities and I-NERI projects related to next-generation reactors and nuclear hydrogen production. KAERI and DOOSAN are members of the GA NGNP team.

The NHDD project shares many common goals with the NGNP Project, and a significant level of collaboration between the projects should be possible in the future. Figure 9-9 shows the overall schedule for the NHDD project, and Figure 9-10 shows and overview of key technologies that are under development. Current activities include development codes and methods for VHTR design, of a bench-scale SI-process facility (see Figure 9-11), development of a high-temperature gas loop for testing SI process components, and development of technologies for compact heat exchangers.

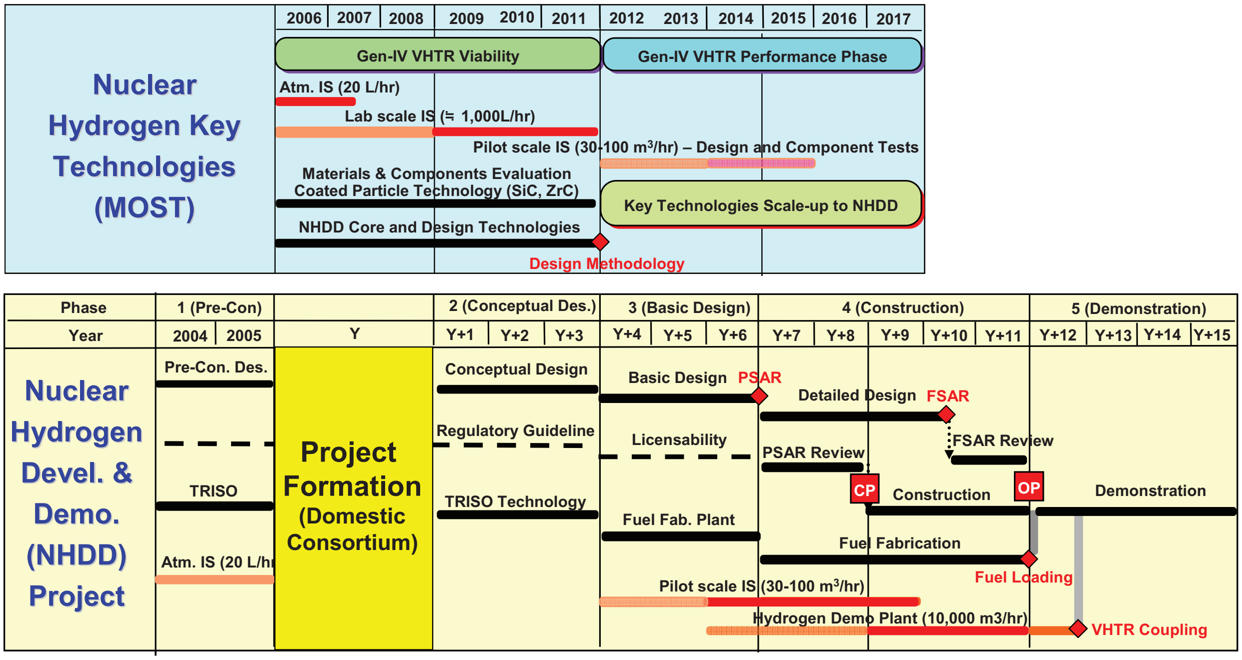


Figure 9-9. NHDD Project Schedule

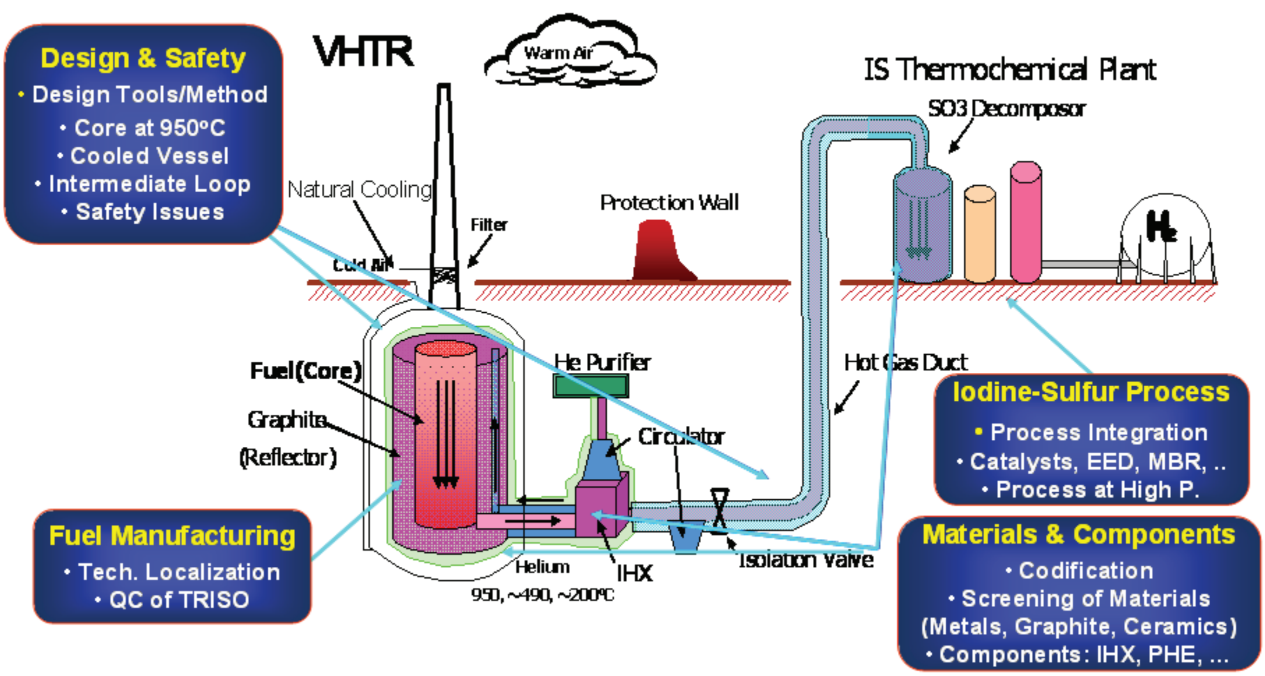


Figure 9-10. Overview of Key Technologies under Development (NHDD Project)

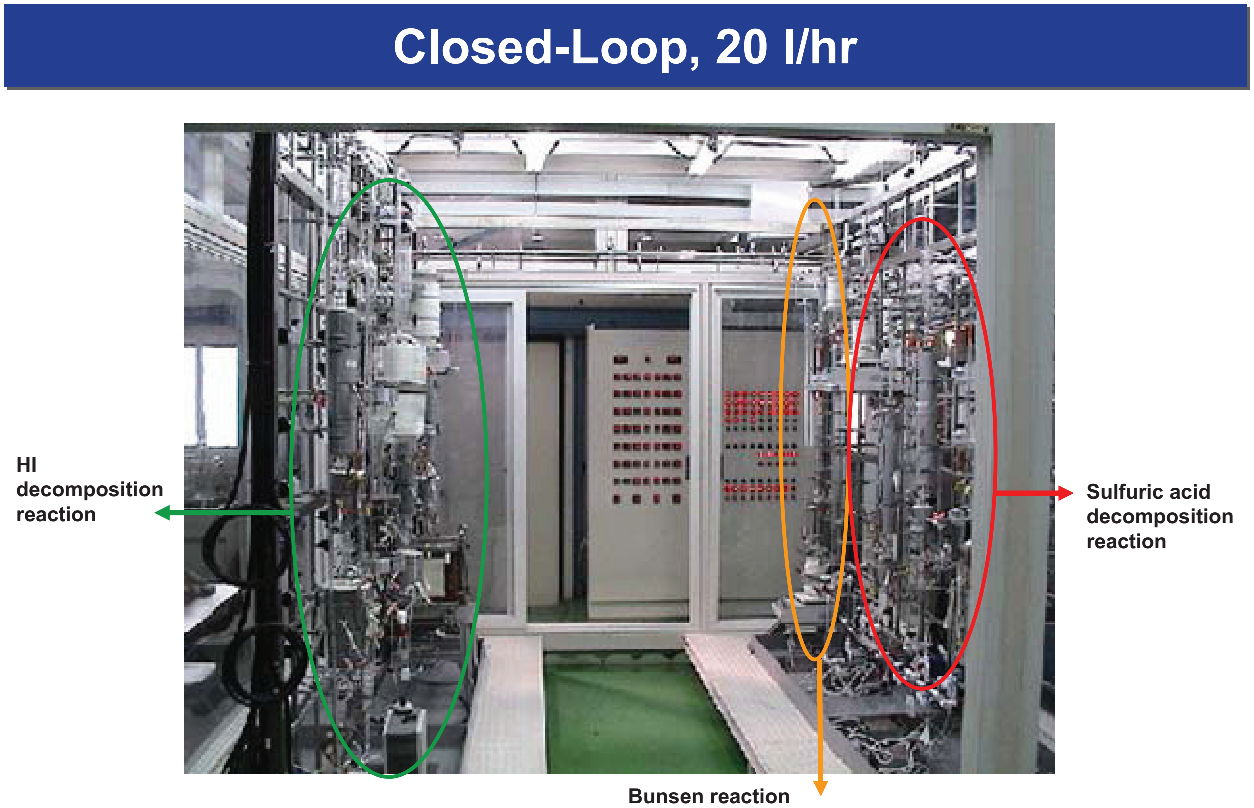


Figure 9-11. Bench-Scale SI Process Facility Located at KIER

9.4 European HTR-TN Program

The High Temperature Reactor Technology Network (HTR-TN) is a European Union (EU)-sanctioned consortium with "…a mission to coordinate and manage expertise and resources required for the development of advanced HTR technologies. It shall assist the European nuclear industry in designing competitive HTR-type power plants with outstanding safety and waste management features including the possibility of burning civil and military plutonium."[51] To accomplish this mission, the EU through the HTR-TN sponsors a number of projects, the most significant of which is RAPHAEL ("Reactor for Process Heat and Electricity") which is comprised of 33 partners from Europe and consists of eight subprograms.[52] The fuel subprogram is described in (Phélip 2006).

The RAPHAEL project will explore the performance of fuel, materials and components, the reactor physics models, the nuclear safety and waste disposal issues, and the overall system integration. This will involve improving the qualification of computer tools and models, exploring the performance limits of fuel and materials, analyzing the behaviour of the fuel under accident and geological disposal conditions, and developing technologies for system components. It will also explore potential interfaces with hydrogen production or process heat exploitation, describing an acceptable nuclear safety approach, and integrating all the results in order to

[51] HTR-TN home page (https://odin.jrc.nl/htr-tn/index.html)
[52] RAEPHAEL home page (http://www.raphael-project.org/index.html)

provide preliminary feasibility assessments of VHTR plants coupled with hydrogen production processes.

In addition to the RAPHAEL project base programs for HTR development, AREVA and CEA are conducting R&D programs in support of the ANTARES design (e.g., Billot 2004, Guillermier 2006).[53]

Obviously, much of this European workscope would be directly relevant to the NGNP Project. At present, there is no non-European participation in HTR-TN or RAPHAEL (Tsinghua University - Institute of Nuclear and New Energy Technology (INET) is attempting to join). The potential for collaboration between NGNP and RAPHAEL is unknown at this time. Based upon past precedent, much of the results will eventually become available through technical conference proceedings and journal articles (e.g., Breuil 2004, Buckthorpe 2004, Fütterer 2006).

Of the European HTR test facilities, the High Flux Reactor (HFR) Petten, NL, is of particular interest.[54] This materials test reactor (MTR) was used extensively by the former German TRISO fuel development program. Consequently, they have fully qualified, multi-capsule test rigs available; however, their on-site capabilities for performing postirradiation examinations of coated-particle fuel is rather limited. The design and operating characteristics of the reactor are especially well suited for fuel irradiation testing (e.g., a properly designed test can operate nearly isothermally to full burnup). A US in-pile fission product transport (HFR-B1) was also irradiated in HFR Petten (Hanson 2006). Arguably, HFR Petten is the premier MTR in the world for coated-particle fuel irradiation testing. It would be available to support the NGNP program on a contract basis.

9.5 PBMR Program

The PBMR Project[55] has planned and is conducting a significant R&D program to support the design and licensing of their prototype pebble-bed module. There does not seem to be an open-literature umbrella TDP for PBMR, but the various elements of their program have been described in a series of papers presented at HTR-2004 (e.g., Fazluddin 2004, Matzner 2004, van der Merwe 2004, etc.) and HTR-2006 (e.g., Hinssen 2006, Müller 2006, Rousseau 2006, etc.). In addition to a fuel fabrication facility, the construction of first-of-a-kind test facilities, including a helium test facility, a heat transfer test facility and a plate-out test facility, is noteworthy.

There are some obvious design differences between PBMR and the NGNP preconceptual design recommended herein (e.g., pebble-bed vs. prismatic cores, oil- vs. magnetic bearings for the PCS, etc.); moreover, the missions are somewhat different with the PBMR dedicated to electricity production, and the NGNP tasked to produce hydrogen as well as electricity. Consequently, there are design-specific DDNs that are different for the two plant designs.

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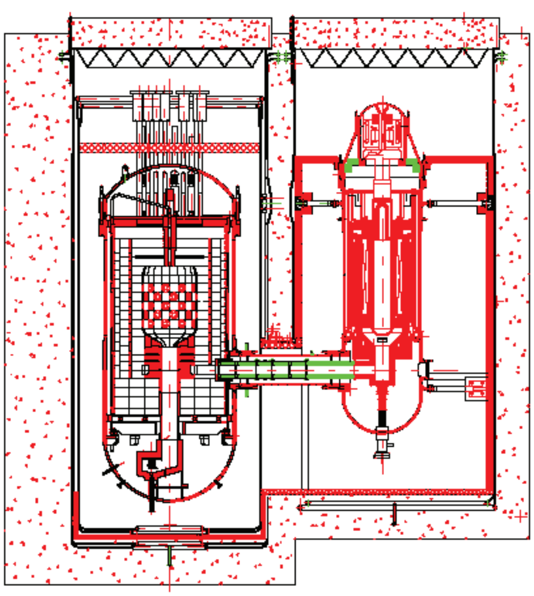
[53] As usual, the interrelationships between the R&D programs sponsored by the EU, by individual countries, and by industry are complex. No attempt will be made here to sort it out.
[54] http://www.jrc.nl/publications/brochures/HFR%20brochure.pdf
[55] http://www.pbmr.com/

Nevertheless, many DDNs, especially those relating to fuel, fission products, graphite, and high-temperature metals, are generic. Technically, there is great potential for collaboration between PBMR and NGNP. As mentioned in the section introduction, the USA and Germany had a very productive cooperative program for gas-cooled reactor development which was not hampered by the fact that the US designs employed prismatic cores and the German designs employed pebble-bed cores. The impediments to collaboration with PBMR appear to be commercial (e.g., intellectual property rights) and political rather than technical. Once the conceptual design has been chosen, the prospects for collaboration should be revisited.

9.6 Chinese MHR Program

The 10 MW(t) High Temperature Gas-cooled Reactor-Test Module (referred to as the HTR-10) has been constructed at the Institute of Nuclear Energy Technology (INET) of Tsinghua University, Beijing, China, to allow the Chinese to develop an expertise in HTGR technology for potential future applications, including electricity production in direct-cycle plants and process heat applications (e.g., Xu 2002). The HTR-10 is a pebble-bed HTR, based upon German plant and fuel technology.

As shown in Figure 9-12, the reactor core and the steam generator are housed in two steel pressure vessels which are arranged side by side. The core power density of the core is 2 MW/m^3. The primary system operates at a pressure of 3.0 MPa and core inlet and outlet temperatures of 250 ^{o}C and 700 ^{o}C, respectively. The core contains 27,000 spherical fuel elements with TRISO-coated, 17% enriched 500-μm UO_2 kernels. The fuel-element design is the same as the reference German design.



Items	Unit	Value
Reactor thermal power	MW	10
Active core volume	m^3	5
Average power density	MW/m^3	2
Primary helium pressure	MPa	3
Helium inlet temperature	^{o}C	250/300
Helium outlet temperature	^{o}C	700/900
Helium mass flow rate	kg/s	4.3/3.2
Fuel		UO_2
U-235 enrichment of fresh fuel elements	%	17
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements		27,000
Refueling mode		Multi-pass, continuously
Average discharge burnup	MWd/t	80,000

Figure 9-12. HTR-10 Design Features

A series of in-reactor safety demonstration tests are currently in progress. INET has announced plans to couple a small gas turbine to the HTR-10 via an IHX (e.g., Huang 2004). The gas turbine has been designed by OKBM and is largely a scaled down version of the PCU design recommended for the NGNP. INET is also investigating the SI process for hydrogen production and the future prospects for coupling an SI pilot plant to the HTR-10.

There does not seem to be an open-literature umbrella TDP for the Chinese HTR program, but various elements of their program have been described in a series of papers given at various IAEA-sponsored meetings on HTGR development. Their planned technology demonstrations would be directly relevant to the NGNP design and licensing; however, it is not clear from the subject papers when these future plans would be implemented. Also, there are no government-to-government enabling agreements to HTGR technology transfer between the US and China.

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10. SCHEDULE AND COST ESTIMATES

Schedule and cost estimates in varying detail are included in the various NGNP and NHI TDPs referred to above, and they are summarized in the NGNP PMPP (2006). In general, it is not possible to critique these schedules and cost estimates with any confidence because the corresponding workscopes are not defined in sufficient detail to permit an independent estimate. This circumstance is especially problematic with the NHI 10-yr R&D Plan for SI and HTE hydrogen process development.

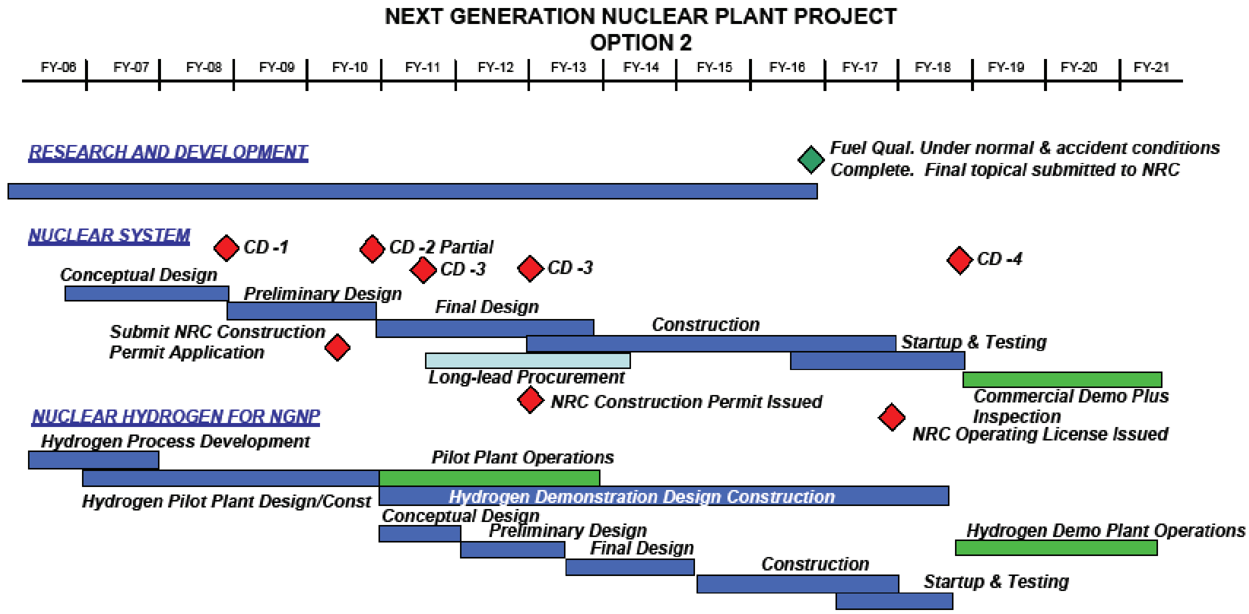
The exception is the AGR fuel development plan wherein the workscope is generally well defined. There are two significant problems with the AGR plan. First, the development schedule is not supportive of the NGNP design and licensing schedule required for a 2018 startup. For example, the Final Design phase would need to be completed by the end of FY2013, but the safety testing and the source term validation tasks are not scheduled for completion until FY2019. The AGR cost estimate appears reasonable for the defined workscope with one notable exception (the cost estimate for constructing an in-pile RN transport loop in the ATR appears to be much too low). However, a more serious problem with the AGR plan is missing workscope which would substantially increase the total program costs: (1) qualification of NFI UO_2 fuel for the initial core and early reloads, (2) an integrated fuel pilot plant to provide the technical basis for a NGNP fuel fabrication facility, and (3) a test program to characterize RN transport in the VLPC.

The PMPP (2006) estimates that the cost for NFI process development and fabrication of AGR-2a test fuel would be ~$6M, and that the cost for irradiation, safety testing and PIE for AGR-2a would be ~$11M. The cost for an integrated fuel fabrication pilot plant would be considerably more and would depend upon the design throughput (e.g., the number of coaters, etc.). As an indication, a fuel fabrication facility with a throughput of 510 fuel elements/year (i.e., a reload segment for a 600 MW(t) NGNP per year) has been estimated to cost ~$200M (PCDSR 2007). The cost for a test program to characterize RN transport in the VLPC cannot be estimated with any confidence at this time because the workscope and experimental approach have not been defined, but it would be not trivial, especially if performed in conjunction with an in-pile loop test program.

10.1 Schedule

The NGNP Option 2 schedule is shown in Figure 10-1 (PMPP 2006). The published schedules for the NGNP and NHI technology development programs are compared to the Option 2 schedule in Figure 10-2.

A comparison of the scheduled completion dates for key deliverables to the dates that the data are needed to NGNP design and licensing (Table 5-8) is provided in Section 11.



Option 2 – Rough Order of Magnitude Estimate Funding Profile

	Prior Year	FY 06	FY 07	FY 08	FY 09	FY 10	FY 11	FY 12	FY 13	FY 14	FY 15	FY 16	FY 17	FY 18	FY 19	FY 20	FY 21
Nuclear System	44	45	100	155	159	198	155	233	284	163	126	146	134	26			
Hydrogen Production	4	25	25	25	29	37	36	32	26	16	17	20	20	5			
Total	48	70	125	180	188	235	190	265	311	180	142	166	154	31			

Figure 10-1. NGNP Deployment Schedule – Option 2

10.2 Cost Estimates

Cost estimates for the various technology programs from FY06 through completion are collected in Table 10-1. The most recent cost estimates for the NGNP and NHI R&D programs are provided in the INL PPMP (2006.

The cost estimate for the PCS was taken from (PCU TDP 2005) which was prepared by OKBM in collaboration with GA and ORNL. This cost estimate was prepared in 2005; consequently, the costs could have been escalated by 6% to 2007$, but that adjustment was not made because it implies a level of confidence in the estimate that is clearly not justified. More to the point, the labor costs at OKBM have increased more rapidly in recent years than the US Consumer Price Index. DOE/ROSATOM would split the costs for the 850 ^{o}C baseline design which would be used for the WPu-burning International GT-MHR, and the NGNP Project would be responsible for incremental costs for the 950 ^{o}C design which has not yet been defined.

The DV&S cost estimate is based upon a 1995 DV&S cost estimate for the commercial GT-MHR escalated to 2007$ (36% increase); the same escalated DV&S costs are included in overall project cost estimate prepared by WGI (PCDSR 2007). Once the conceptual design of the NGNP is defined, these DV&S cost estimates should be revisited. This cost estimate does not include critical experiments to validate nuclear design methods (Section 3.10.1). If critical experiments are confirmed to be required, their costs will need to be added. Also, these DV&S

tasks are scheduled to be completed over a 3-yr time period in 2011 – 2013; a 5-6 year timeframe seems more realistic for these tests.

The existing cost estimates total $1,029,130, and certain cost elements are clearly missing. The major omissions along with preliminary cost estimates are: (1) qualification of NFI UO_2 fuel for the initial core (~$17M not including the cost of the reactor fuel); modification of the OKBM PCS design for 950 °C (~$25M, 25% of the 850 °C base case cost); and (3) nuclear critical experiments (undetermined at this writing). In addition, the Energy Transfer TDP, when it is prepared, may contain additional component DV&S tests (e.g., large isolation valves for the Secondary Heat Transport System, high temperature circulators, etc.).

The cost estimate in the PPMP includes $176M for thermochemical hydrogen R&D and $135M for HTE hydrogen R&D; these are substantial monies, but the workscopes for the hydrogen tasks so lack specificity that it is difficult to judge the appropriateness of these cost estimates.

Taken at face value, the existing cost estimates imply that the NGNP technology development programs urgently need to be reprioritized. Arguably, the highest priority technology task is the qualification of UCO fuel and establishment of the technical basis for the design and construction of a domestic fuel fabrication facility. However, only 13% of the total cost estimate is for fuel qualification. Even more striking, only 3% of the total is for validating the radionuclide source terms which will be essential for licensing.

The cost estimates for metals R&D ($150M) and, to a somewhat lesser extent, for graphite R&D ($83M) are also excessive. Clearly, the number of candidate metals and graphites to be investigated need to be reduced, and the qualification tests prioritized. As previously recommended, the metals programs should focus on 2¼Cr-1Mo and 9Cr-1Mo-V for the RPV and on IN 617 and a backup (e.g., Hastelloy XR) for the IHX. The graphite program should focus on qualifying a replacement graphite for H-451. The materials R&D cost estimate includes $12.4M for "…IHX Fabrication, Testing and Evaluation…." Presumably, this task is an IHX DV&S task, and the qualification testing of candidate IHX materials (e.g., IN 617) is included in the various metals cost elements. If a new large-scale He test loop needs to be constructed for thee IHX DV&S tests, the actual cost could be significantly higher.

The methods development and validation cost estimate is excessive at $106M (11% of the total), especially since it is focused on nuclear and thermal/fluid dynamic methods. As discussed previously (Section 3.10), industry standard codes, such as MCNP and ANSYS, are already routinely used for design and analysis of prismatic MHRs. This cost estimate includes $40M for a large-scale mockup of the Reactor System for validation of thermal/fluid flow predictive methods for normal operation and accidents. This facility is not needed for a prismatic Reactor System. Some small-scale mockup tests might be prudent (e.g., to characterize hot streaks in the hot duct), but these DV&S tests would be design specific, and their need cannot be determined until late in the Preliminary Design phase. Moreover, integral validation data will be generated during the hot-flow tests (prior to initial criticality) during the startup and commissioning phase.

The task of focusing and prioritizing the technology programs will become more straightforward once the NGNP design is officially determined. At that time, a stand-alone, bottoms-up TDP needs to be prepared which is responsive to the NGNP DDNs and the design and licensing schedule. Presumably, the total R&D costs can be reduced significantly.

10.3 References for Section 10

[PCDSR] "NGNP and Hydrogen Production Preconceptual Design Studies Report," GA Document 911107, Rev. 0, General Atomics, July 2007.

[PCU TDP] PCU "Technology Demonstration Program Plan, Revision 2005 (Draft)," Product No: 08.03-006.01A, OKBM, 2005 *(Business Confidential).*

[PPMP] Weaver, K., et al., "Next Generation Nuclear Plant Project Preliminary Project Management Plan," INL/EXT-05-00952, Rev. 1, Idaho National Laboratory, March 2006.

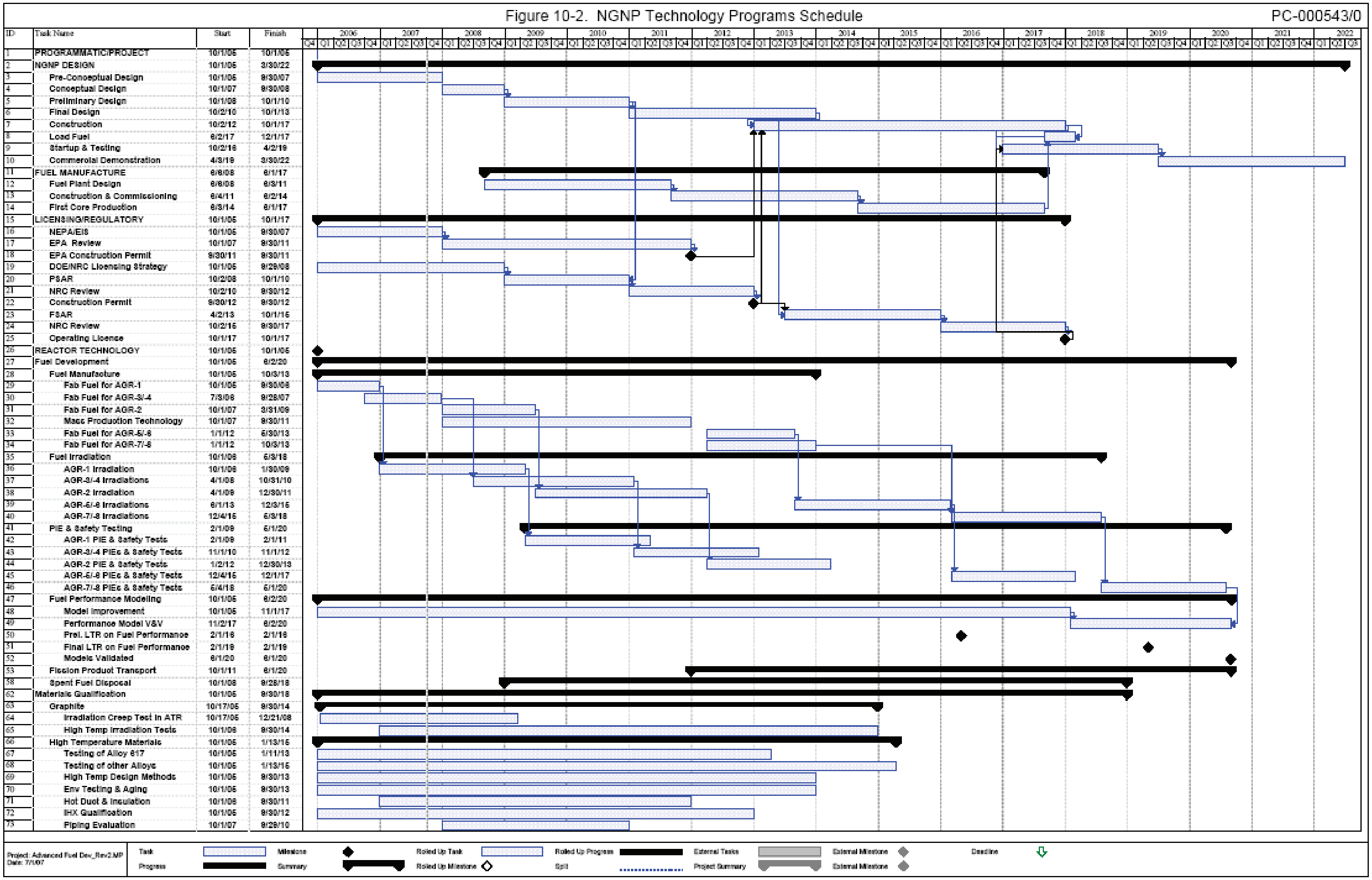
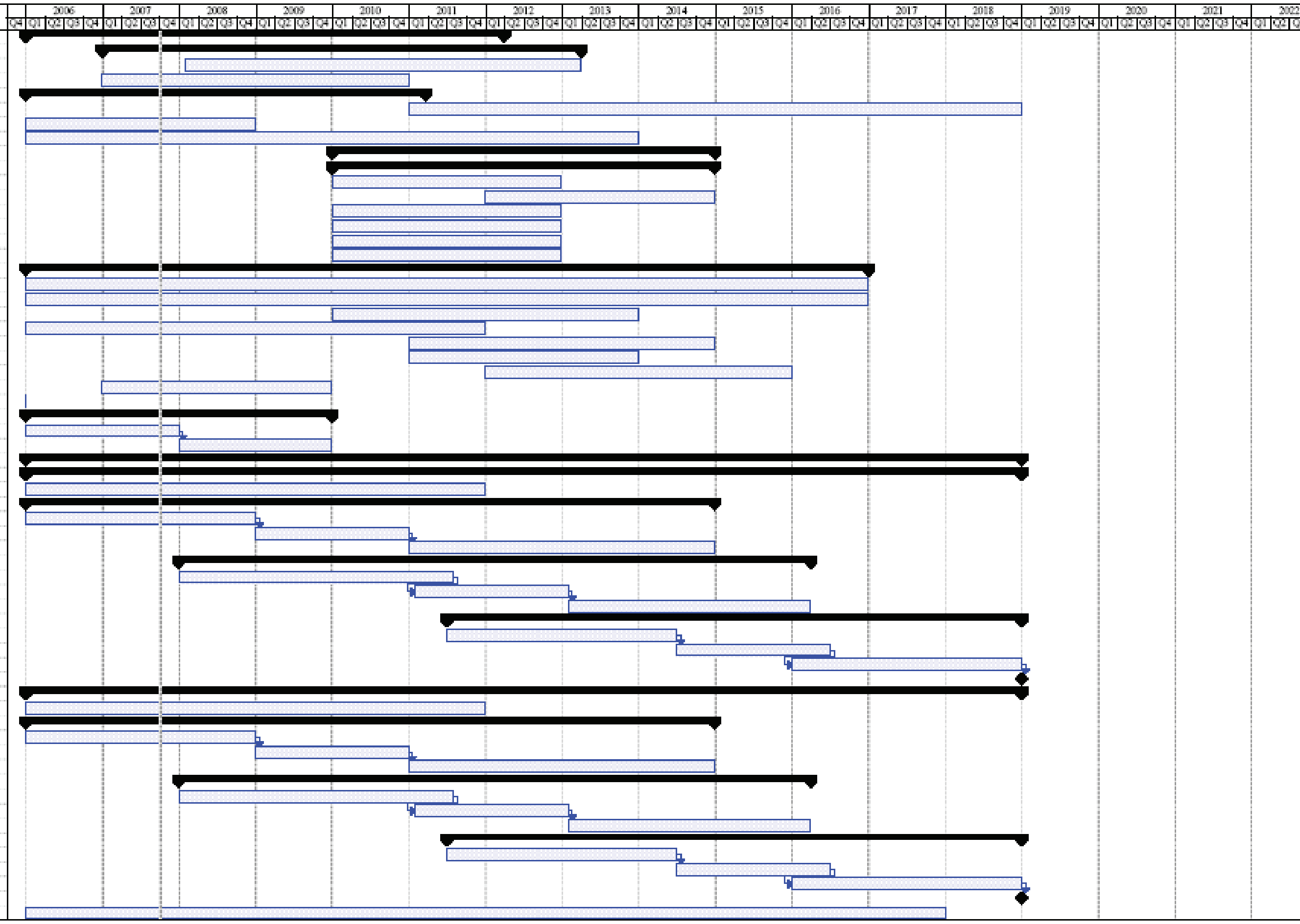


Figure 10-2. NGNP Technology Programs Schedule

Figure 10-2. NGNP Technology Programs Schedule

PC-000543/0

ID	Task Name	Start	Finish
74	Code Case Support	10/1/05	12/30/11
78	RVP Material Testing	10/1/06	12/31/12
79	Irradiation of RPV Materials	10/30/07	12/31/12
80	RPV Emissivity	10/1/06	9/30/10
81	Composites Development	10/1/05	12/17/10
84	Const & Ops Support	10/1/10	9/30/18
85	Facility Upgrades	10/1/05	9/29/08
86	Administration	10/1/05	9/30/13
87	DV&S Programs	10/1/09	9/29/14
88	Reactor System	10/1/09	9/29/14
89	Critical Experiments	10/1/09	9/28/12
90	Component DV&S	10/1/11	9/29/14
91	Vessel System DV&S	10/1/09	9/28/12
92	RCCS DV&S	10/1/09	9/28/12
93	FH&SS DV&S	10/1/09	9/28/12
94	PCD&IS DV&S	10/1/09	9/28/12
95	Power Conversion System	10/1/05	9/30/16
96	PCU Integration	10/1/05	9/30/16
97	Full-Scale Turbocompressor	10/1/05	9/30/16
98	Generator	10/1/09	9/30/13
99	Electromagnetic Bearings	10/1/05	10/1/11
100	Recuperator	10/1/10	9/30/14
101	Precooler-Intercooler	10/1/10	9/28/13
102	PCU IVM	10/1/11	9/30/15
103	Alternate PCU: Dry Gas Seal	10/1/06	9/29/09
104	HYDROGEN TECHNOLOGY	10/1/05	10/1/05
105	Systems, BOP	10/1/05	9/29/09
106	IHX Design, Materials Selection	10/1/05	9/30/07
107	IHX Development for PP	10/1/07	9/29/09
108	H2 Production Technologies	10/1/05	10/1/18
109	Thermochemical (SI) Process	10/1/05	10/1/18
110	Lab-scale Development	10/1/05	9/30/11
111	SI Pilot-Scale Development	10/1/05	9/30/14
112	Pilot-Plant Design	10/1/05	9/30/08
113	Pilot Plant Construction	10/1/08	9/30/10
114	Pilot Plant Operation	10/1/10	9/30/14
115	SI 6-MW Pilot-Plant Dev.	10/1/07	12/31/16
116	5-MW Pilot-Plant Design	10/1/07	6/2/11
117	5-MW Pilot Plant Constructio	11/1/10	10/30/12
118	5-MW Pilot Plant Operation	10/31/12	12/31/16
119	~1/10 Scale SI Demo Plant	4/1/11	10/1/18
120	Preliminary Design	4/1/11	3/31/14
121	Final Design	4/1/14	3/31/16
122	Construction	10/2/15	10/1/18
123	Begin Operation	10/1/18	10/1/18
124	HTE Process	10/1/05	10/1/18
125	Lab-scale Development	10/1/05	9/30/11
126	HTE Pilot-Plant Development	10/1/05	9/30/14
127	Pilot-Plant Design	10/1/05	9/30/08
128	Pilot Plant Construction	10/1/08	9/30/10
129	Pilot Plant Operation	10/1/10	9/30/14
130	HTE 6-MW Pilot-Plant Dev.	10/1/07	12/31/16
131	5-MW Pilot-Plant Design	10/1/07	6/2/11
132	5-MW Pilot Plant Constructio	11/1/10	10/30/12
133	5-MW Pilot Plant Operation	10/31/12	12/31/16
134	~1/10 HTE Scale Demo Plant	4/1/11	10/1/18
135	Preliminary Design	4/1/11	3/31/14
136	Final Design	4/1/14	3/31/16
137	Construction	10/2/15	10/1/18
138	Begin Operation	10/1/18	10/1/18
139	H2 Systems Development	10/1/05	9/30/17



Project: Advanced Fuel Dev_Rev2.MP
Date: 7/1/07

Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Milestone | Rolled Up Progress | Split | External Tasks | Project Summary | External Milestone | External Milestone | Deadline

Table 10-1. NGNP Technology Development Cost Estimates

Technology Area	NGNP Technology Development Cost Estimates ($ X1000)														
	2006	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**	**2013**	**2014**	**2015**	**2016**	**2017**	**2018**	**2019**	**Total**
Fuel Dev.[a]	12,525	13,575	13,200	12,825	13,150	11,350	8,000	7,300	6,900	3,975	9725	6750	6,700	8,025	134,000
RN Transport[a,b]	75	200	250	250	250	400	3,700	3,500	5,200	8,400	3,300	650	1,700	2,600	30,475
Graphite[c]	3,700	10,000	14,900	18,000	16,000	13,200	5,000	2,000	500						83,300
Metals[c,d]	7,500	18,200	26,000	32,000	23,900	12,800	7,300	5,700	3,500	3,500	3,500	3,000	3,000		149,900
IHX[c,e]	700	2,000	4,300	2,000	2,000	1,000	400								12,400
PCS (OKBM)[f]	1,963	7,978	7,452	3,394	8,544	15,070	25,132	16,506	8,364	4,982	7,409				106,794
DV&S[g]						31,400	31,400	31,400							94,200
H2 – SI[c]	15,100	15,100	15,500	17,002	20,605	22,621	20,091	15,223	8,852	7,326	8,618	8,378	2,054		176,470
H2 – HTE[c]	9,200	9,900	9,500	11,692	16,419	13,197	12,225	11,136	7,470	9,293	11,414	11,294	2,754		135,494
Methods[c]	6,290	6,668	11,009	11,089	10,039	10,039	10,231	8,783	8,783	7,722	7,722	7,722			106,097
Total	**57,053**	**83,621**	**102,111**	**108,252**	**110,907**	**131,077**	**123,479**	**101,548**	**49,569**	**45,198**	**51,688**	**37,794**	**16,208**	**10,625**	**1,029,130**

[a] AGR Fuel Development Plant, Rev. 1, 2005 (unescalated).

[b] The irradiation, PIE and safety testing cost of three fission product release tests are included in the Fuel Development subtotal; they add another ~$20M to the total for RN transport.

[c] INL PPMP, 2006 (unescalated).

[d] All materials R&D costs except for graphite and IHX component tests are included here.

[e] This cost element is just for the IHX component tests (NGNP Materials R&D Plan 2005); IHX metals qualification included in "Metals" cost element.

[f] Cost estimate from (PCU TDP 2005) (unescalated). 50/50 cost sharing by DOE/ROSATOM for 850 ^{o}C design; NGNP Project would be responsible for incremental cost for 950 ^{o}C design which has not yet been defined.

[g] GA 1995 DV&S cost estimate for commercial GT-MHR escalated to 2007; same DV&S costs included in overall project cost estimate prepared by WGI (PCDSR 2007).

11. KEY DELIVERABLES

The NGNP DDNs were described in Section 5, including the required delivery dates for data production in order to support design and licensing (Table 5-8); these required dates are relational rather than calendar dates that tied to major program events (e.g., the start of Final Design, prior to plant startup, etc.). The PMPP Option 2 schedule, including a 2018 plant startup, was assumed for translating the relational DDN delivery dates into approximate calendar dates (Section 10.1). The planned delivery dates for the various technology programs (Section 6) are given as calendar dates; for the NGNP and NHI technology programs, the most recent planned delivery dates are given in the NGNP PMPP (2006) which was issued in March 2006.

The planned production dates for key deliverables are compared to the required dates in Table 11-1. As previously discussed, a number of the technology development schedules do not support the Option 2 NGNP design and licensing schedule. Resolution of this disconnect represents a major programmatic challenge to senior NGNP Project management which needs to be addressed expeditiously.

Some of the delivery dates given in Table 5-8 are later in the design process than called in the GT-MHR DDNs (1996). For example, component DV&S data are generally called for prior to completion of Final Design. For the GT-MHR, some of these DV&S DDNs call for the data at the end of Preliminary Design (i.e., three years earlier). These DDNs were delayed with the rationale that the design of a FOAK plant would have to progress beyond a two-year Preliminary Design before the SSCs could be sufficiently well defined to define the test specifications for a DV&S test program. Nevertheless, a strong case could be made that these DV&S results are needed at least one year prior to completion of Final Design. The need dates for each NGNP DDN should be systematically reevaluated during Conceptual Design.

11.1 Comparison of Planned and Required Delivery Dates

The planned and required production dates for key deliverables are compared below by technology area. The level of detail varies from area to area. Consequently, the reliability of evaluation varies correspondingly. For example, the AGR Fuel Plan/1 (2005) is quite detailed and quite quantitative. Other TDPs, such the as DV&S TDPs, have not yet been prepared.

Fuel Development and Qualification

The largest schedule disconnect is between NGNP design and licensing schedule and the planned AGR fuel program schedule. Only AGR-1 irradiation and postirradiation heating data will be available at the start of Final Design. The PIE and safety testing of the UCO qualification fuel (AGR-5/-6) will not be completed until four years after Final Design is complete. The fuel performance models that are critical to making the safety case will not be completely validated until almost three years after the Operating License is needed from the NRC to support a 2018 startup.

The disconnect between the design and licensing schedule and the AGR fuel schedule is so great that the GA team has recommended the use of NFI UO_2 fuel for the initial core and the first one or two reloads (PCDSR 2007). This recommendation was made reluctantly because the GA team also strongly believes that qualified UCO fuel, including a domestic fuel supplier, is essential for commercial viability of the H2-MHR.

Radionuclide Transport

Validation of the radionuclide source term, which is deemed critical to obtaining an Operating License from the NRC, is part of the AGR fuel program and also suffers from the same large schedule disconnect. Fission product release data for UCO fuel under irradiation and during postirradiation heating tests (AGR-3/-4)[56] are scheduled to be available one year prior to completion of Final Design. The in-pile loop tests to characterize plateout in the primary coolant circuit and liftoff during rapid depressurization accidents will not be completed until three years after Final Design and a year after the FSAR is submitted (Figure 10-2). In fact, the feasibility of constructing such an in-pile loop in the ATR is not scheduled to be investigated until FY13 (the last year of Final Design). Characterization of RN transport in the VLPC during core heatup accidents (Section 3.2.2.3), and tritium transport (Section 3.2.2.4) are not currently addressed in the AGR Fuel Plan. Finally, the NGNP source terms will not be completely validated until almost three years after the Operating License is needed from the NRC to support a 2018 startup.

The PPMP concluded that the planned RN transport program may not be adequate to validate the NGNP source terms, and the GA team concurs. The total planned RN transport budget is only $30M (~$50M when the irradiation, PIE and safety testing costs of three fission product release tests are included); moreover, the planned annual expenditures do not exceed $400K until FY12. The workscope related to RN transport in the primary circuit (e.g., sorption measurements, loop tests, etc.) should be accelerated, and workscope for RN transport in the VLPC and tritium transport should be added.

Materials Qualification

The level of detail provided for the materials qualification program is much less than that provided for the fuel and fission product programs. With that caveat, the planned delivery dates for the key data appear responsive to the NGNP design schedule. The RPV and IHX metals are codified by the start of Final Design, and the core graphites are codified a year later. However, the metals and graphite characterization tests both continue another three years after codification. The RPV emissivity tests are complete by the start of Final Design.

Energy Transfer Technology

The energy transfer TDP has not yet been prepared. The only available information was gleaned from the PPMP (2006) and the NGNP Materials R&D Plan (2005). The PPMP indicates that IHX testing will be completed in 10/12 which is a year before the completion of Final Design. Given that the IHX is arguably the highest risk component in the NGNP, it would

[56] The design of AGR-3/-4 is currently about one year behind schedule due to severe underfunding of the AGR program in FY07.

be highly desirable to have these data two years earlier at the start of Final Design. Presumably, these IHX tests will have a major influence on the final choice of heat exchanger design for the NGNP (PCHE versus helical coil). Once the energy transfer TDP is prepared, the planned delivery dates for key test data should be reviewed.

Power Conversion System

The planned OKBM PCU development schedule (PCU TDP 2005) is marginally responsive to the NGNP Option 2 design and licensing schedule. The critically important EMB tests are completed two years to the end of Final Design. The recuperator tests start one year into Final Design and are largely complete by the end of Final Design. The full-scale turbocompressor tests continue until three years beyond the completion of Final Design although the annual expenditures ramp down considerably during those final three years, implying that the bulk of the TC data will be available by the end of Final Design. The incremental testing to support upgrading the OKBM machine for operation at 950 ^{o}C has not yet been scheduled, and that program planning should be completed expeditiously.

Given the complexity of this FOAK machine and the concerns raised by various independent reviewers, including Rolls-Royce (Appendix C, PCDSR 2007), it would be highly desirable for these demonstration tests to be accelerated. Moreover, the timely development of a backup design (e.g., the RR-recommended combined cycle) should also be planned.

Design Verification and Support

The various DV&S TDPs have also not yet been prepared. The data are required prior to the completion of Final Design. The only planning done to date was in the context of developing an overall NGNP cost estimate (Section 5, PCDSR 2007). For that purpose, it was assumed that the previous DV&S costs for the NGNP nuclear heat source would be comparable to the 1995 DV&S cost estimate for the commercial GT-MHR (with escalation). It was further assumed that the DV&S testing would be completed during a three-year period corresponding to Final Design.

DV&S TDPs should be prepared early during Preliminary Design, and an effort should be made to spread testing over a longer time period. Some testing with small-scale mockups might be performed during the last year of Preliminary Design, and some DV&S testing will be done on large-scale prototypes (e.g., circulators) and/or early production units of actual reactor components (e.g., control rods). Some of the latter confirmatory testing could be done after formal completion of Final Design.

Hydrogen Production

The planned development programs for both the SI-based and the HTE-based hydrogen production technologies are predicated upon the same logic (NHI 2005): lab-scale tests leading to a small-scale pilot plant followed by an engineering-scale (5 MW) pilot plant and finally a ~1/10-scale demonstration plant coupled to the NGNP. Given the immature state of both hydrogen technologies, it is critically important that the data from the small-scale- and engineering-scale pilot plants are available to support the design of the FOAK NGNP demonstration plants. The schedules for the hydrogen programs, which are virtually identical, are taken from Figure 4-2 in PPMP which is somewhat different from the NGNP Option 2 schedule from the same document (reproduced here as Figure 10-1).

The small-scale pilot plants begin operation in FY10, and the testing programs are finished four years later. The final designs of the 5-MW pilot plants are completed six months after the startup of the small-scale pilot plants so the latter will only provide limited data to support the design of the former. Neither pilot plant will have generated significant data by the time the Preliminary Design of 1/10-scale NGNP demonstration plants begin in mid-FY11. Both pilot-plant programs should be accelerated so that the data from the small-scale pilot plants are available no later than the start of Final Design for NGNP hydrogen plant (4/14) and data from the 5-MW pilot plants are available no later than one year prior to completion of the Final Design (4/15).

Spent Fuel Disposal

As previously discussed, technology development to support spent fuel disposal for the NGNP and follow-on commercial H2-MHRs is not included in the PPMP. A spent fuel disposal TDP has been prepared (Hanson 2002). It should be independently reviewed, revised as necessary and included in the PPMP.

Methods Development and Validation

The GA perspective on the design methods development and validation program was discussed at some length in Section 6.7. Those misgivings aside, several planned delivery dates are too late to support the Option 2 NGNP design and licensing schedule. The V&V of thermal/fluid dynamics methods needs to be completed at least 1 year earlier (1 yr prior to the scheduled date for obtaining an Operating License). The neutronics and thermal/fluid dynamics methods really should be validated one prior to completion of Final Design (10/13); given that industry standard codes are being used, this goal appears to be achievable.

A RCCS DV&S test program is included in the current methods development plan. Design of the test facility (an existing facility Argonne is to be modified) would begin in FY08 and the testing would be complete at the end of FY15. The RCCS test facility needs to reflect the specific NGNP design. Proper design of the test facility cannot be assured until Preliminary Design is completed at the end of FY10. Given that the RCCS is an essential system for maintaining fuel temperatures below 1600 ^{o}C during core heatup accidents and for protecting the RPV, the RCCS DV&S tests should be complete by the end of Final Design (two years earlier than currently scheduled).

11.2 References for Section 11

[AGR Plan/1] “Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program” INL/EXT-05-00465, Rev. 1, Idaho National Laboratory, August 2005.

[GT-MHR DDNs] “600 MW(t) Gas Turbine Modular Helium Reactor Design Data Needs,” DOE-GT-MHR-100217, General Atomics, July 1996.

Hanson, D. L., and M. B. Richards, “[Commercial GT-MHR] Spent Fuel Disposal Confirmatory Test and Analysis Plan,” PC-000503, Rev. 0, General Atomics, June 2002.

[NGNP Materials R&D Plan] "Next Generation Nuclear Plant Materials Research and Development Program Plan," INEEL/EXT-05-00758, Rev. 2, Idaho National Laboratory, September 2005.

[NHI Plan] "Nuclear Hydrogen Initiative Ten Year Program Plan," US Department of Energy, March 2005.

[PCDSR] "NGNP and Hydrogen Production Preconceptual Design Studies Report," GA Document 911107, Rev. 0, General Atomics, July 2007.

[PCU TDP] PCU "Technology Demonstration Program Plan, Revision 2005 (Draft)," Product No: 08.03-006.01A, OKBM, 2005 *(Business Confidential).*

[PPMP] Weaver, K., et al., "Next Generation Nuclear Plant Project Preliminary Project Management Plan," INL/EXT-05-00952, Rev. 1, Idaho National Laboratory, March 2006.

Table 11-1. Key Deliverables for Technology Programs

Deliverable	Completion Date		Programmatic Basis for Required Date
	Scheduled	Required	
Fuel Development ("Scheduled" dates from AGR Fuel Plan/1)			
AGR-1 irradiation complete	01/09	10/09	1yr prior to completion of Preliminary Design
AGR-1 PIE/safety tests complete	02/11	10/09	1yr prior to completion of Preliminary Design
AGR-2 irradiation complete	01/12	10/10	Start of Final Design
AGR-2 PIE/safety tests complete	01/14	10/10	Start of Final Design
Preliminary NRC topical report on fuel performance	02/16	10/10	In conjunction with PSAR submittal
AGR-5/-6 irradiations complete	12/15	10/13	Prior to completion of Final Design
AGR-5/-6 PIE/safety tests complete	12/17	10/13	Prior to completion of Final Design
Final NRC topical report on fuel performance	01/19	10/15	In conjunction with FSAR submittal
AGR-7 irradiations complete	05/18	10/16	1yr prior to Operating License
AGR-7 PIE/safety tests complete	05/20	10/16	1yr prior to Operating License
Fuel performance models validated	06/20	10/16	1yr prior to Operating License
Radionuclide Transport ("Scheduled" dates from AGR Fuel Plan/1)			
AGR-3/-4 irradiations complete	11/10	10/10	Start of Final Design
AGR-3/-4 PIE/safety tests complete	10/12	10/10	Start of Final Design
In-pile loop tests complete	10/16	10/16	1yr prior to Operating License
AGR-8 irradiation complete	05/18	10/16	1yr prior to Operating License
AGR-8 PIE/safety tests complete	05/20	10/16	1yr prior to Operating License
RN transport in VLPC tests complete	TBD	10/16	1yr prior to Operating License
NGNP source terms validated	05/20	10/16	1yr prior to Operating License
Materials Qualification ("Scheduled" dates from PPMP)			
Graphites and composites codified	10/11	10/12	1 yr prior to completion of Final Design
Graphite testing complete	10/14	10/13	Prior to completion of Final Design
RPV and IHX metals codified	07/10	10/10	Start of Final Design
Environmental tests complete	10/13	10/13	Prior to completion of Final Design

Deliverable	Completion Date		Programmatic Basis for Required Date
	Scheduled	Required	
RPV irradiation tests complete	10/13	10/13	Prior to completion of Final Design
RPV emissivity tests complete	10/10	10/10	Start of Final Design
Energy Transfer Technology ("Scheduled" dates from PPMP)			
Hot duct/insulation tests complete	10/11	10/10	Start of Final Design
IHX tests complete	10/12	10/10	Start of Final Design
Power Conversion System ("Scheduled" dates from PCU TDP)			
Turbocompressor tests complete	12/16	10/13	Prior to completion of Final Design
Generator tests complete	12/13	10/13	Prior to completion of Final Design
EMB tests complete	12/11	10/10	Start of Final Design
Recuperator tests complete	12/14	10/13	Prior to completion of Final Design
Precooler/intercooler tests complete	12/13	10/13	Prior to completion of Final Design
Dry gas seal tests complete (alternate design concept)	12/09	10/10	Start of Final Design
PCU integration tests complete	12/16	10/13	Prior to completion of Final Design
Design Verification & Support ("Scheduled" dates from PCDSR cost estimate)			
Nuclear Heat Source DV&S tests complete	10/13	10/13	Prior to completion of Final Design
Hydrogen Production – SI ("Scheduled" dates from PPMP)			
SI integrated Loop Construction Completion	09/07	09/07	Must be complete prior to closed-loop operation
SI Integrated Loop Test Results	09/08	09/08	Required for Pilot-Plant design
Complete small SI pilot-plant construction	10/10		Prerequisite for test program
Complete small SI pilot-plant experiments	10/14	04/14	Start of Final Design of NGNP SI plant
Complete 5 MW SI pilot-plant construction	05/11		Prerequisite for test program
Complete 5 MW SI pilot-plant experiments	01/16	04/15	1 yr prior to completion of Final Design of NGNP SI plant
Hydrogen Production – HTE ("Scheduled" dates from PPMP)			
Select HTE pilot-plant process	10/08	10/07	1yr prior to completion of pilot plant Final Design
Complete HTE lab-scale tests	10/11	10/12	1yr prior to completion of Final

Deliverable	Completion Date		Programmatic Basis for Required Date
	Scheduled	Required	
			Design of NGNP SI plant
Complete small HTE pilot-plant construction	10/10		Prerequisite for test program
Complete small HTE pilot-plant experiments	10/14	04/14	Start of Final Design of NGNP HTE plant
Complete 5 MW HTE pilot-plant construction	05/11		Prerequisite for test program
Complete 5 MW HTE pilot-plant experiments	01/16	04/15	1 yr prior to completion of Final Design of NGNP HTE plant
Spent Fuel Disposal (not included in PPMP)			
Complete graphite “noncombustibility” demonstration	TBD	10/09	1yr prior to completion of Preliminary Design
Complete long-term particle leaching tests	TBD	10/17	Spent fuel disposition determined prior to Operating License
Complete C-14 production and transport characterization	TBD	10/17	Spent fuel disposition determined prior to Operating License
Methods Development and Validation (“Scheduled” dates from PPMP)			
Neutronics methods validated	07/12	10/16	1yr prior to Operating License
Thermal/fluids methods validated	10/17	10/16	1yr prior to Operating License
RCCS tests complete	10/15	10/13	Prior to completion of Final Design
Integral VHTR tests complete	10/17	N/A	Not needed

APPENDIX A. DESIGN DATA NEEDS

The following DDN template is based upon the DDN format and content used for the commercial GT-MHR;* these DDNs evolved over many years and became progressively more detailed in the process. As an example of a "mature" DDN, DDN C.01.01.04, "Quality Control Test Techniques Development," is also included here (an early version of this generic fuel QC DDN was first prepared for the steam-cycle MHTGR in 1986). Initial issues of new DDNs (e.g., SI DDN N.44.04.01, "Corrosion Performance") would likely be less detailed.

* "DOE Projects Division Program Directive #16: HTGR PROGRAMS - Design Data Needs (DDNs) Interim Procedure," PD#16, Rev. 1, February 1986.

DATE:

NGNP PROJECT
[DDN Title]
DDN [N.XX.YY.ZZ]

PLANT: NGNP/System [number]

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

[Brief statement providing context (e.g., what is the significance or impact of the needed data on the design) and providing a justification for the DDN.]

1.1 Summary of Functions/Assumptions

[from Systems Requirements Manual]

1.2 Current Data Base Summary

[brief summary; several key references as available]

1.3 Data Needed

[Succinct statement of data needed; specify required accuracy if appropriate]

Quality assurance must be adequate to meet requirements for components which will be classified as [specify].

1.4 Data Parameters/Test Conditions

Parameter (as applicable)	Value
[materials, test articles, etc.]	
[e.g., temperature range]	
[e.g., pressure range]	
[e.g., fast fluence range]	
[e.g., chemistry]	
[e.g., duration]	
[etc.]	

2. DESIGNER'S ALTERNATIVES

Alternatives to the proposed approach are as follows:

2.1 Alternative 1

2.2 Alternative 2, etc.

3. SELECTED DESIGN APPROACH AND EXPLANATION

[Describe why each alternative given in Section 2 was rejected.]

4. SCHEDULE REQUIREMENTS

Preliminary data: [specify]; final data [specify]. [Tie to design phases rather than calendar dates.]

5. PRIORITY

[specify: high/medium/low] [relative to the importance of new data]

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

[State credible fallback position if no new data are generated and programmatic consequences**]**

7. REFERENCES

[Include as available]

[specify: name, organization]
Originator Date

[specify: name, organization]
Engineering Manager Date

[specify: name, organization]
Project Manager Date

DATE: 6/30/94

GT-MHR PROGRAM
QUALITY CONTROL TEST TECHNIQUES DEVELOPMENT
DDN C.01.01.04

PLANT: GT-MHR/Multi-System 07

1. REQUIREMENT OR DESIGN FEATURE REQUIRING EXPERIMENTAL DATA OR VALIDATION TESTING

The fuel for the GT-MHR must have low levels of as-manufactured defects, as well as properties that ensure the structural integrity of the coated particles will be maintained during irradiation. Improved QC techniques are needed to demonstrate that the GT-MHR fuel will meet the stringent quality requirements with high confidence. Further, the QC techniques must be automated to improve the reproducibility and decrease the time required for measurements, consistent with the production plant requirements.

1.1 Summary of Function/Assumption

"Control Radionuclide Transport from Core," Assumption: Processes are available for manufacturing high quality fuel kernels, coatings and compacts for inclusion in prismatic fuel elements.

1.2 Current Data Base Summary

At present, the QC techniques available for inspection and testing of GT-MHR fuel components are essentially the methods used for inspection of Fort St. Vrain (FSV) fuel (Ref. 1). Many of these methods employ technologies which are inherently time consuming and labor intensive. Although adequate for inspection of small quantities of fuel in support of fuel development activities, these techniques require improvement for use in a fuel production facility. The extent of automation of most of the existing methods is minimal currently. One measurement which has been automated is the examination of coated fuel particle batches for missing buffer layers, as determined by image analysis of x-ray photos.

The sensitivity of the burn-leach test, which is the primary method to measure defects in the SiC coating layers, is limited by gas and liquid transport through pores in the coatings. As the quality of the fuel is improved, the relative contribution from smaller defects becomes more significant. A need has been demonstrated for a more sensitive technique than burn-leach to measure the levels of SiC defect fractions in fuel compacts.

The program in Germany for the development of TRISO coated fuel particles in pebble elements for the High Temperature Reactor included techniques to characterize the fuel quality (Ref. 2). This work included the development of techniques to measure the microstructure of SiC and PyC on developmental fuel. Similar developmental methods have been continued in Japan in support of the High Temperature Test Reactor.

The examination in the US of irradiated fuel samples which failed during the MHTGR and NPR capsule irradiation tests has shown that some product attributes not previously measured need to be characterized more completely. These attributes are the SiC strength, SiC microstructure, PyC microstructure and PyC permeability. Fuel meeting the low as-manufactured defect fractions of the GT-MHR fuel has been previously manufactured, but much of this fuel has performed poorly during irradiation testing.

The evaluation of the fuel failure in past irradiation capsule tests has indicated that the improvement in fuel performance required for the GT-MHR must come primarily from fundamental improvements in the fuel product and process specifications. However, improved QC techniques will also be needed to demonstrate compliance with the fuel specifications.

1.3 Data Needed

Qualified and documented procedures are needed for characterizing the fuel using improved inspection techniques, including the following specific approaches:

1. More efficient methods are needed to perform measurements which are now performed manually, including automated image analysis measurement techniques for the fraction of particles with fuel dispersion, the fraction with missing coatings, the coating thicknesses, the carbide content of individual fuel kernels, and for dimensional and surface condition inspection of fuel compacts.

2. A more sensitive SiC defect detection method is needed than the burn-leach technique, which needs to be capable of measuring defects of less than one micron size.

3. Techniques are needed for detection of non spatial SiC defects (e.g., low local strength, poor microstructure or internal flaws), the candidates for which include a SiC strength test, optical measurements and x-ray measurements.

4. An improved method is needed for directly measuring the permeability of the IPyC coating layer, for which the candidates include gaseous HCl leaching and methylene iodide intrusion.

5. An improved method is needed for characterizing the microstructures of the IPyC and the OPyC layers, the results of which can be correlated with irradiation performance.

6. A method is needed for quantitatively measuring the matrix intrusion into the OPyC coatings within fuel compacts.

Quality assurance must be in accordance with the requirements for experimental data or validation testing which is "safety related".

1.4 Data Parameters/Service Conditions

The qualified procedures for fuel characterization must be capable of measuring the following product attributes to the requirements of the fuel product specification (Ref. 3):

Quality Requirements	Fraction Fissile or Fertile Mean, at 95% Confidence	95% Conf. ≤5% of Compacts Exceed
Missing or defective SiC	$\leq 5 \times 10^{-5}$	1×10^{-4}
Heavy metal contamination	$\leq 1 \times 10^{-5}$	2×10^{-5}
Total fraction HM outside SiC	$\leq 6 \times 10^{-5}$	1.2×10^{-5}
Missing or defective IPyC	$\leq 4 \times 10^{-5}$	1×10^{-4}
Missing or defective buffer layer	$\leq 5 \times 10^{-5}$	1×10^{-4}

2. DESIGNER'S ALTERNATIVES

The alternative to the proposed approach is to use existing techniques which were largely developed for characterizing Fort St. Vrain fuel. The alternative approach would require a greater reliance on process controls to ensure that the fuel product will meet the product specifications.

3. SELECTED DESIGN APPROACH AND EXPLANATION

The selected approach is to procure new equipment as required and to develop QC technique improvements for the characterization of coated particle fuels in compacted bodies. Documentation will be prepared for qualifying the improved equipment, and detailed procedures will be written for performing all QC tests. Fuel samples for irradiation tests will be inspected using the improved techniques.

The alternative for characterization techniques was not selected because it would result in a higher risk of fuel failure during irradiation. The selected approach is based on utilizing product-based specifications to the greatest extent achievable, as opposed to process-based specifications for controlling product properties.

The selection of more automated techniques was made to reduce the inspection delay and the costs for QC measurements in a production facility. The automation development can be achieved after the characterization of fuel for qualification tests.

4. SCHEDULE REQUIREMENTS

The improved characterization techniques shall be in place to assure that they can be utilized in time to characterize the fuel for qualification irradiation tests. The procedures for characterizing test fuel must be completed three months prior to the start of fabrication of the qualification test fuel. The improved automated image analysis techniques are needed three months prior to the proof testing of fuel.

5. PRIORITY

Priorities, as defined in Reference 4, are:

Urgency: 1
Cost benefit: H
Uncertainty in existing data: H
Importance of new data: H

6. FALLBACK POSITION AND CONSEQUENCES OF NONEXECUTION

The fallback position would be to use the techniques developed for the Fort St. Vrain fuel production, along with some improvements made in fuel process controls. The consequence of this action would be to increase the risk that the fuel would not be adequate to meet the performance requirements, with a probable increase in radionuclide releases from a reactor core. The implication for the reactor design is that a pressurized secondary containment may become necessary.

7. REFERENCES

1. "MHTGR Fuel Process and Quality Control Description," DOE-HTGR-90257, September 1991.

2. Nabielek, H., et al., "Development of Advanced HTR Fuel Elements," Nuclear Engineering & Design, 1990.

3. Fuel Product Specification for GT-MHR," DOE-GT-MHR-100209, May 1994.

____
Originator Date

____
Engineering Manager Date

____
Project Manager Date

APPENDIX B. SUMMARY OF OKBM PCU TDP

The following viewgraph presentation provides a pictorial summary of the extensive technology program currently in progress in Russia to qualify the OKBM PCU design.

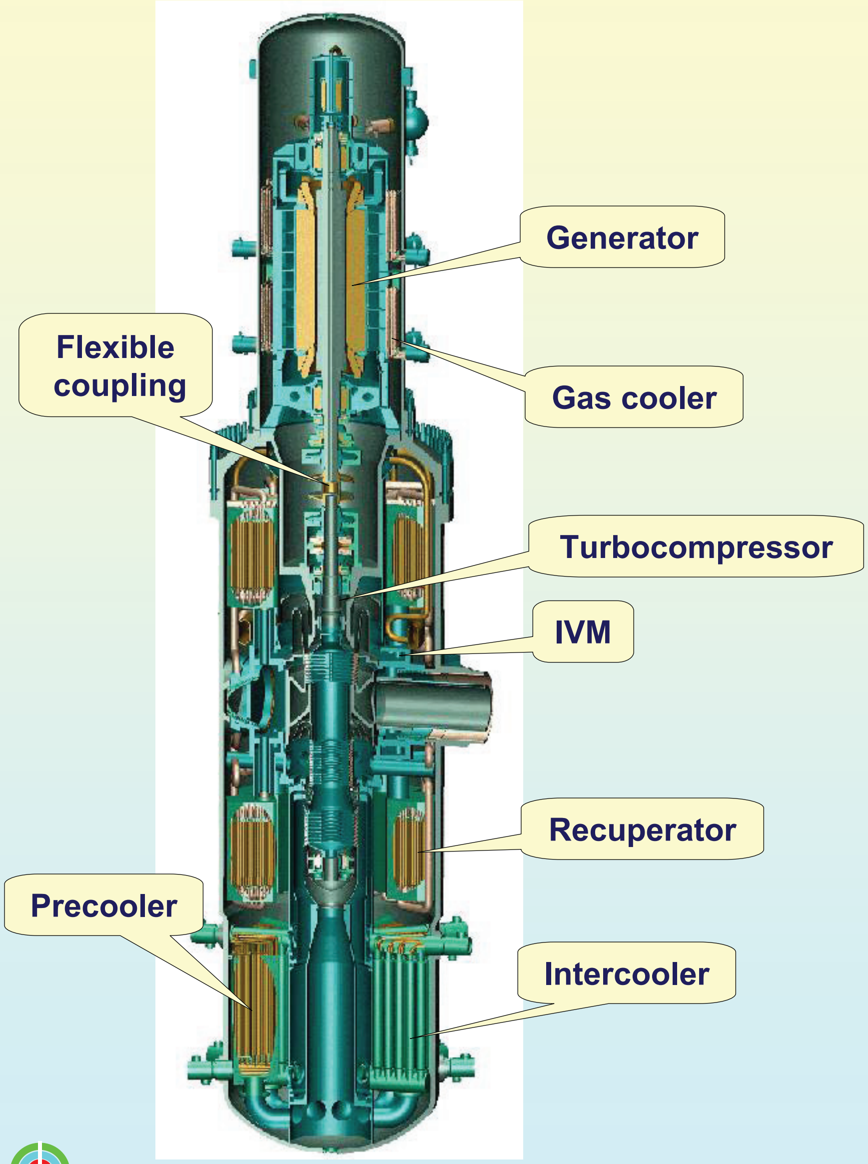
GT-MHR International project.
PCU

July, 2005

ОКБМ

Part I.
PCU Technology Development Program

GT-MHR PCU Concept



Key Features of Reference PCU for Development Plan

- *Integrated layout of PCU equipment inside a single vessel*
- *Vertical arrangement of Turbomachine*
- *Flexible coupling between rotors of generator and TC*
- *Sliding seals of TC stator*
- *Electromagnetic Bearing Support*
- *Helium cooling of Generator*
- *Modular Prime Surface Recuperators*
- *Recuperator effectivness – 0,95*
- *Modular Precoolers and Intercoolers*

Main technical characteristics

600 MWt/287MWe	
Generator power, MW	*287*
Helium temperature at PCU inlet, ^{0}C	*850*
Helium temperature at PCU outlet, ^{0}C	*490*
Helium pressure at PCU inlet, MPa	*7.03*
Helium pressure at PCU outlet, MPa	*7.12*
4400 rpm TM	

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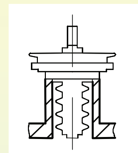
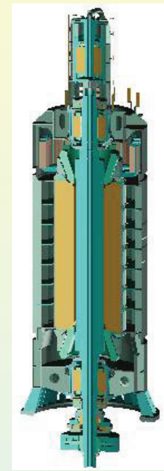
PCU Technology Development Program

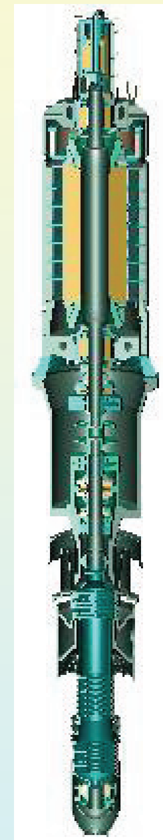
Electric Lead-Outs



TC, Generator



Turbomachine



PCU

- **Subcomponent Tests**
- **Component Tests**
- **TM testing at plant site**
- **Start-up testing at first plant**

PCU Development Program Based on Four-Phase Strategy

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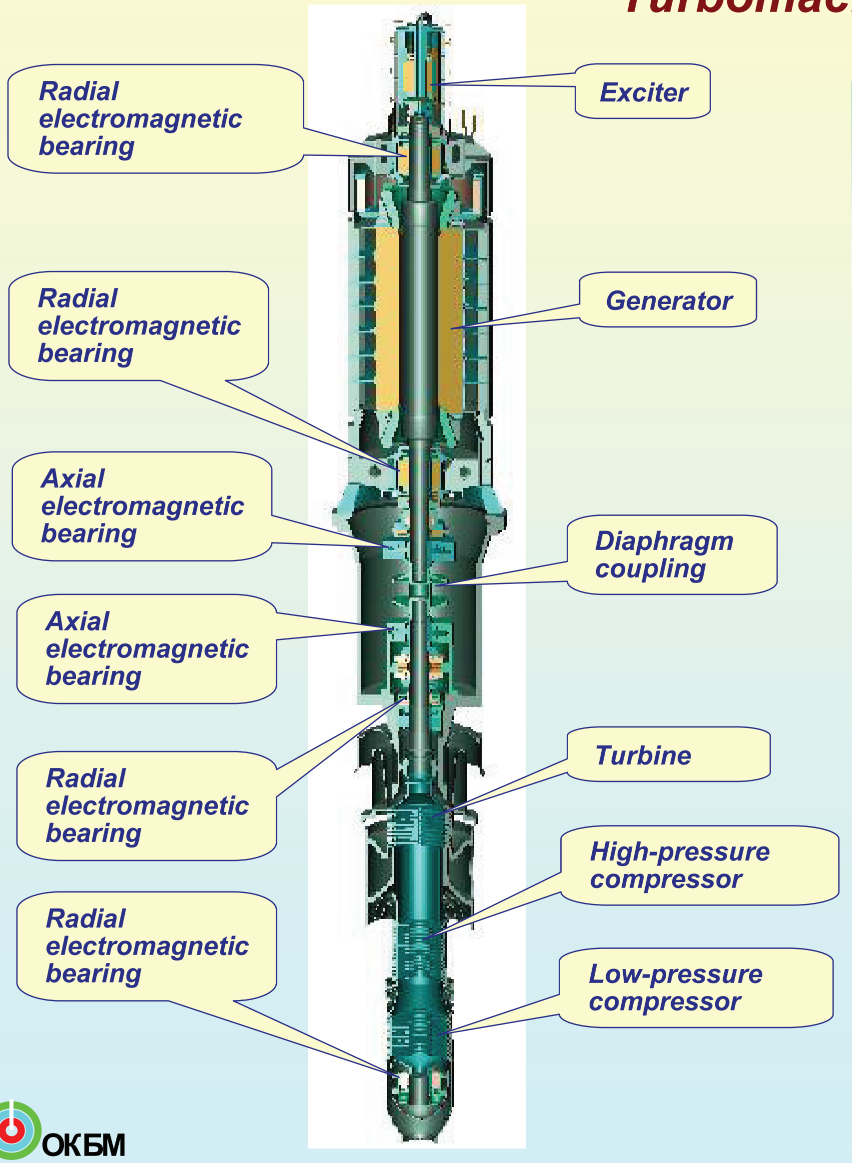
PCU Technology Development Program

Turbomachine



TM functions

- TM is intended to convert thermal power into electrical power in a direct gas-turbine cycle with high-temperature gas-cooled reactor

Basic performance characteristics

Generator power, MW	287
Rotation speed, rpm	4400

Design features

- Operating fluid – helium
- Vertical arrangement
- Helium-cooled generator
- Rotor on a full electromagnetic suspension
- Sliding seals of turbocompressor (TC) stator
- Diaphragm coupling between generator and TC rotors
- Leak-tight electrical penetrations through PCU vessel

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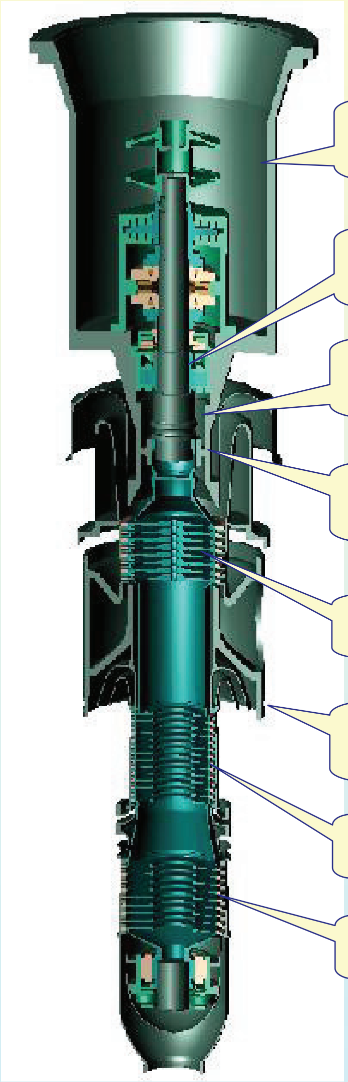
PCU Technology Development Program

Turbocompressor



Turbocompressor functions

Turbocompressor is intended to convert thermal power into mechanical power, provide coolant circulation in RP primary circuit and drive the generator

Basic performance characteristics

❑ Turbine power, MW	558.5
❑ Helium parameters at turbine inlet	
• Pressure, MPa	7.03
• Temperature, ^{0}C	850
• Flowrate, kg/s	322
❑ Rotation speed, rpm	4400
❑ Number of stages	
• Turbine	9
• LPC	13
• HPC	10
❑ Rotor length, m	~ 13.5
❑ Rotor mass, t	~ 30

ОКБМ

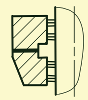
PCU Technology Development Program

Development Program

First Plant

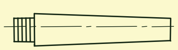
- **TC Rotor Sealing**

Verification of TC rotor brush and labyrinth seal design in helium



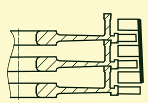
- **Turbine Materials**

Search, selection and qualification of HT structural materials



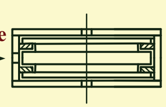
- **LPC**

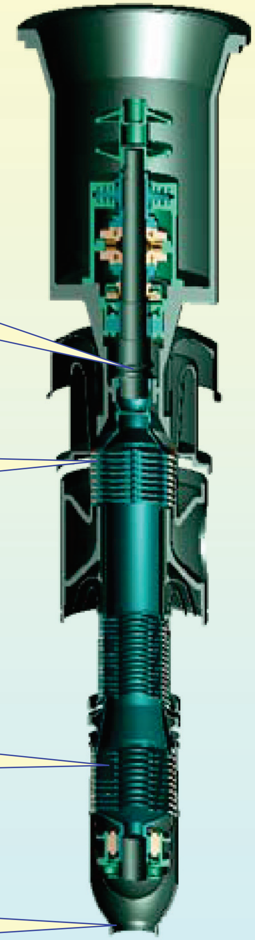
Verification of the low pressure compressor aero design in helium



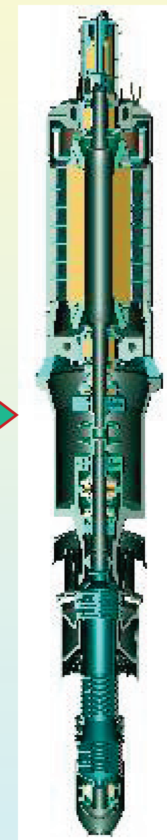
- **TC Stator Seals**

Verification of TC Stator seals design in air and helium





Full-scale turbocompressor tests



TM testing at plant site

Start-up testing at first plant

Turbocompressor Development Logic

ОКБМ

PCU Technology Development Program

Test goal

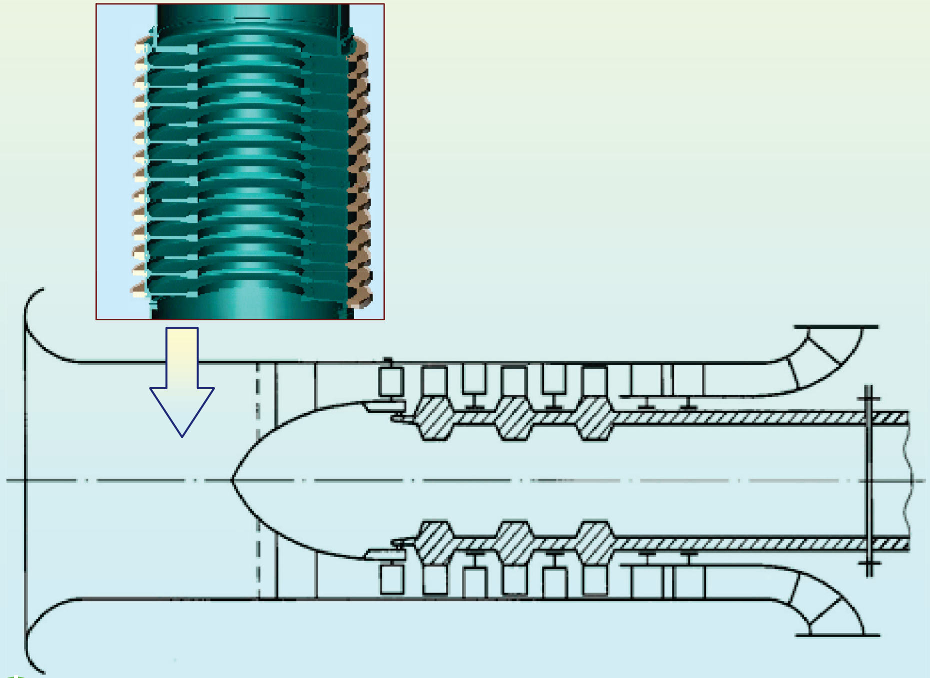
- Verify procedures to design and profile compressor blades for operation in helium fluid
- Investigate stage effectiveness
- Define gas-dynamic characteristics
- Define pressure loss in supply and removal channels

Test facility characteristics

Working fluid	helium
Rotor speed, rpm	3200-26000
Number of stages	3
Simulation factor	0.204
Consumed power, kW	750
Inlet temperature, °C	20-30
Inlet pressure, MPa	0.1-0.588

Status

- ✓ Technical assignment for test facility has been developed
- ✓ Test program has been developed
- ✓ Test facility flow diagram has been developed



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Model Compressor Testing

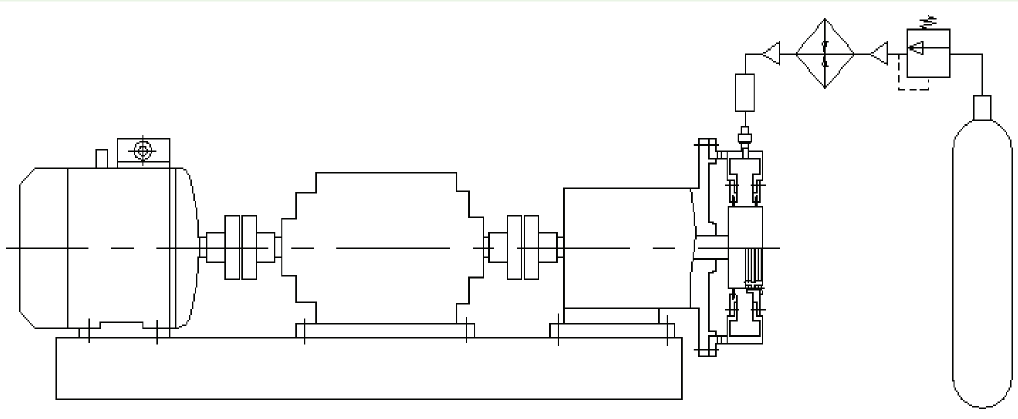
PCU Technology Development Program

Test goal

- Define leakage across the seals as a function of pressure drops and alignment of surfaces to be sealed
- Investigate the effect of radial gaps in the seals on leakage amount
- Check operability and reliability

Test facility characteristics

Working fluid	helium
Rotor speed, rpm	42000
Consumed power, kW	5
Inlet temperature, °C	110-850
Inlet pressure drops, MPa	0.58



Test facility for TC rotor brush and labyrinth seals

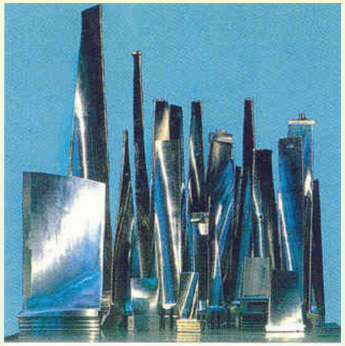
Status

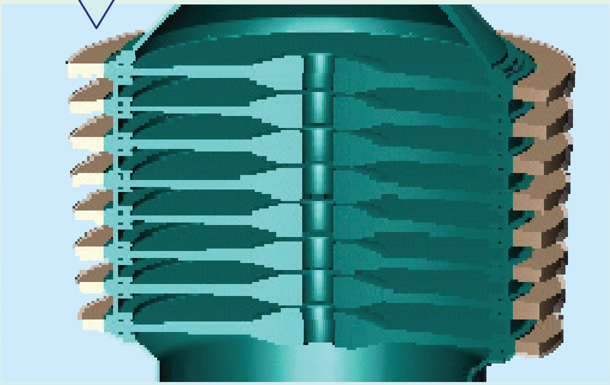
- ✓ Technical assignment for test facility has been developed
- ✓ Test program has been developed
- ✓ Test facility flow diagram has been developed

TC Rotor Brush and Labyrinth Seals Testing

ОКБМ

PCU Technology Development Program





Test conditions

- Working fluid — air/helium
- Temperature, °C — 20-850

Status

Candidate materials for high-temperature turbine elements have been selected:

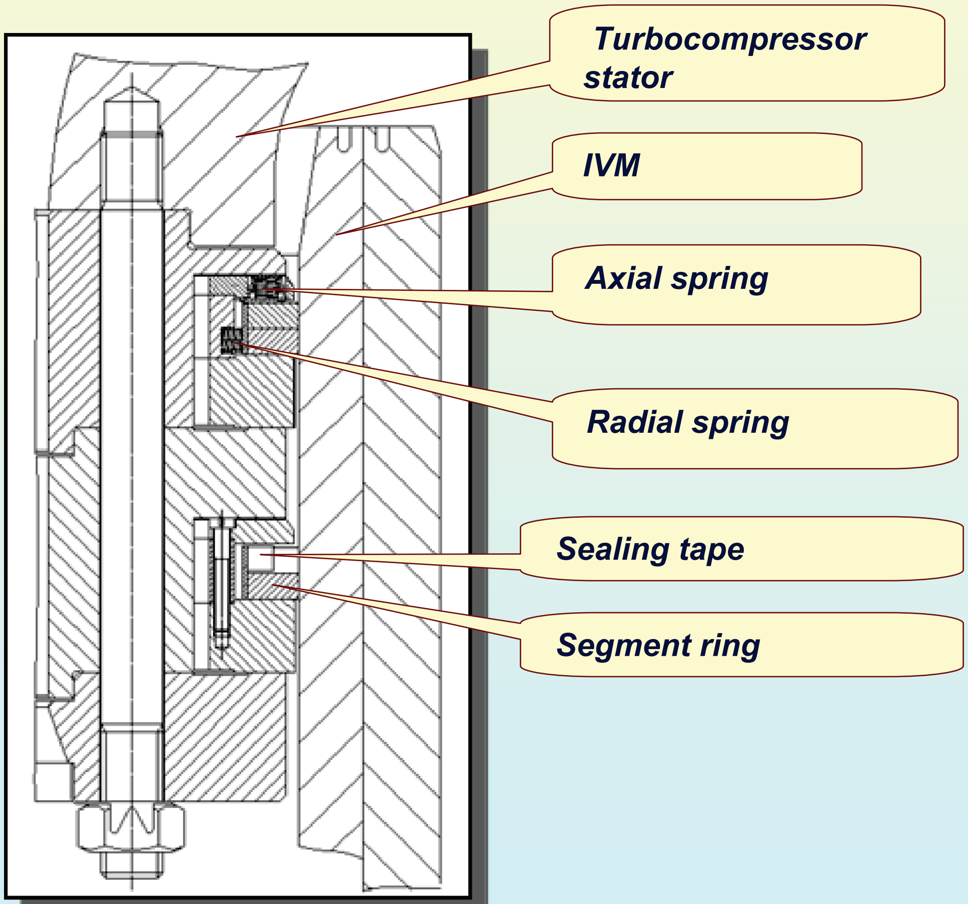
- Discs — CrNi73MoNbTiAl (EI 698-VD)
- Blades — ZhS 6 alloy
- Stator — CrNi55MoWZr (ChS 57)

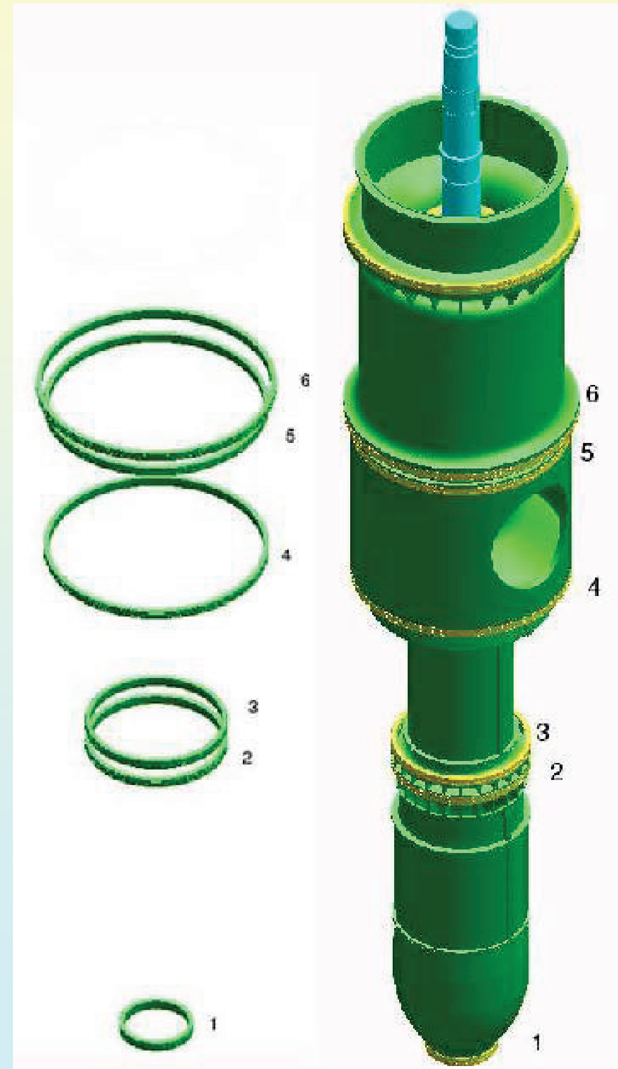
Turbine Materials

ОКБМ

PCU Technology Development Program

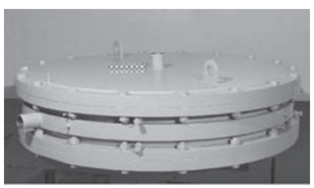
Stator Seals





PCU Technology Development Program

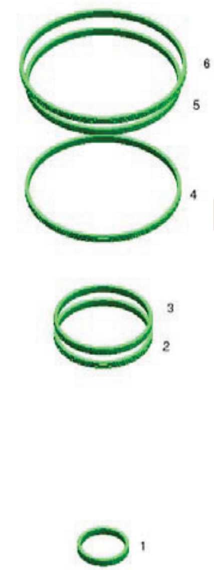
Cold tests of seal mockup in air

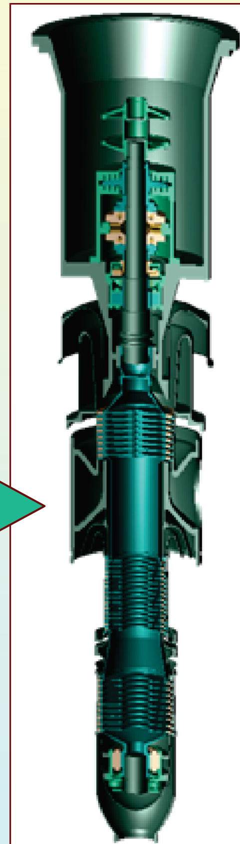


Cold tests of pilot seal sample in air and helium

Hot tests of pilot seal sample in helium (6 modes)

Cold tests of 6 standard sizes of stator seals in helium





Development Program on Turbocompressor Sliding Stator Seals

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PCU Technology Development Program

Investigation of mockups of turbocompressor stator seals in air

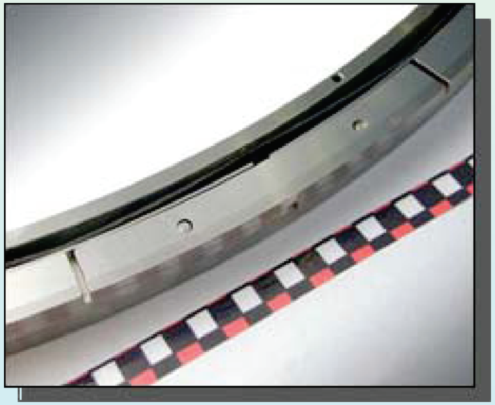
ОКБМ

PCU Technology Development Program

Stator Seal Ring

- Limits helium leaks
- Allows radial and axial displacement of the turbocompressor
- Provides radial friction support

Stator seal basic elements



Yoke of stator seal segments

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PCU Technology Development Program

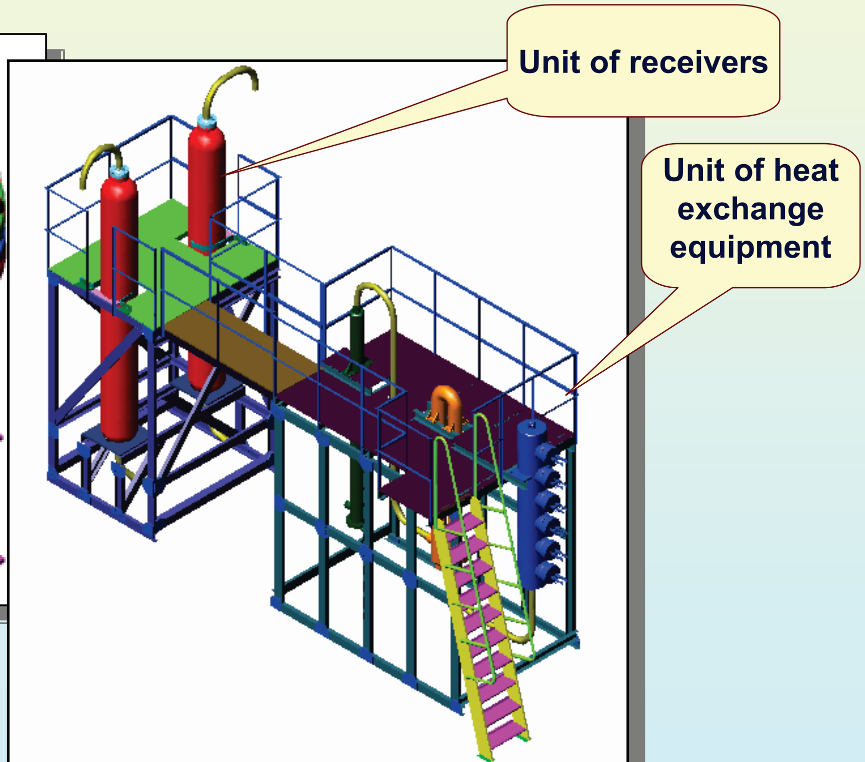
Test parameters

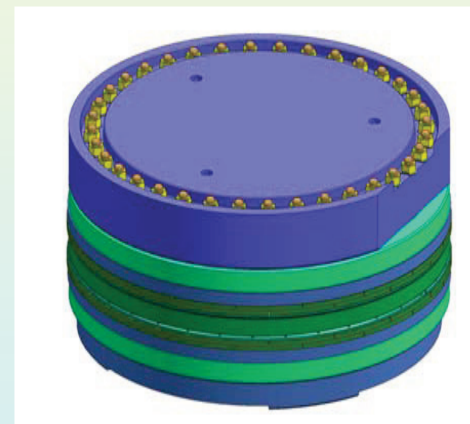
- Working fluid: air – helium
- Temperature - to 50 – 500 °C
- Pressure - to 7.6 MPa

Test Objective

Determine values of air and helium leaks through the seal depending on pressure drop in it and different temperatures and positions of seal internal casing

Pilot sample of TC stator seal

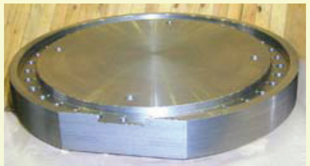


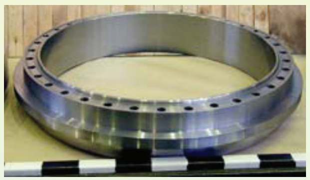


Removable part of TC stator seal pilot sample for tests in helium

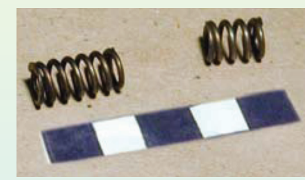
PCU Technology Development Program

Removable part of TC stator seal pilot sample for tests in helium



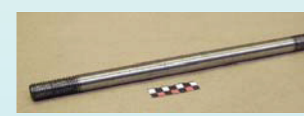




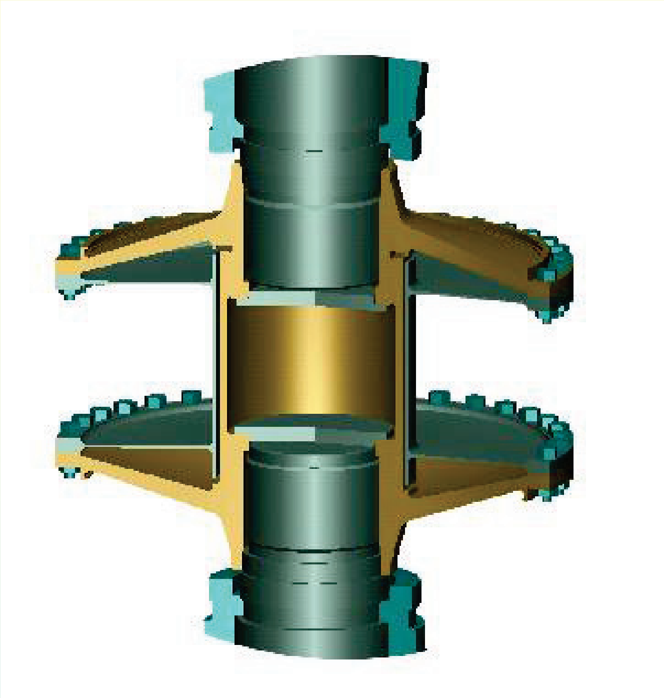


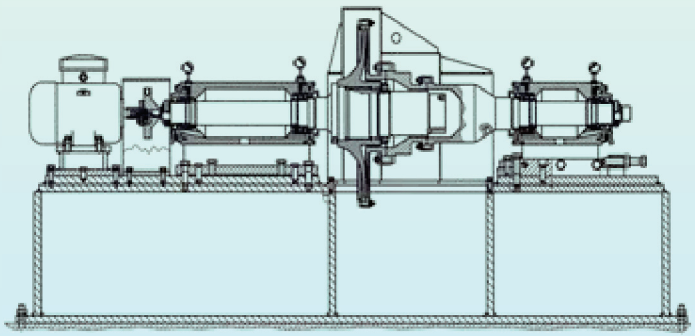




PCU Technology Development Program





Test goals

- Investigate coupling stress-strained state due to axial, angular displacement of the shaft and torque without rotation
- Investigate coupling strength due to cyclic effect of the torque
- Investigate coupling strength under multicycle loading during rotation

Test facility characteristics

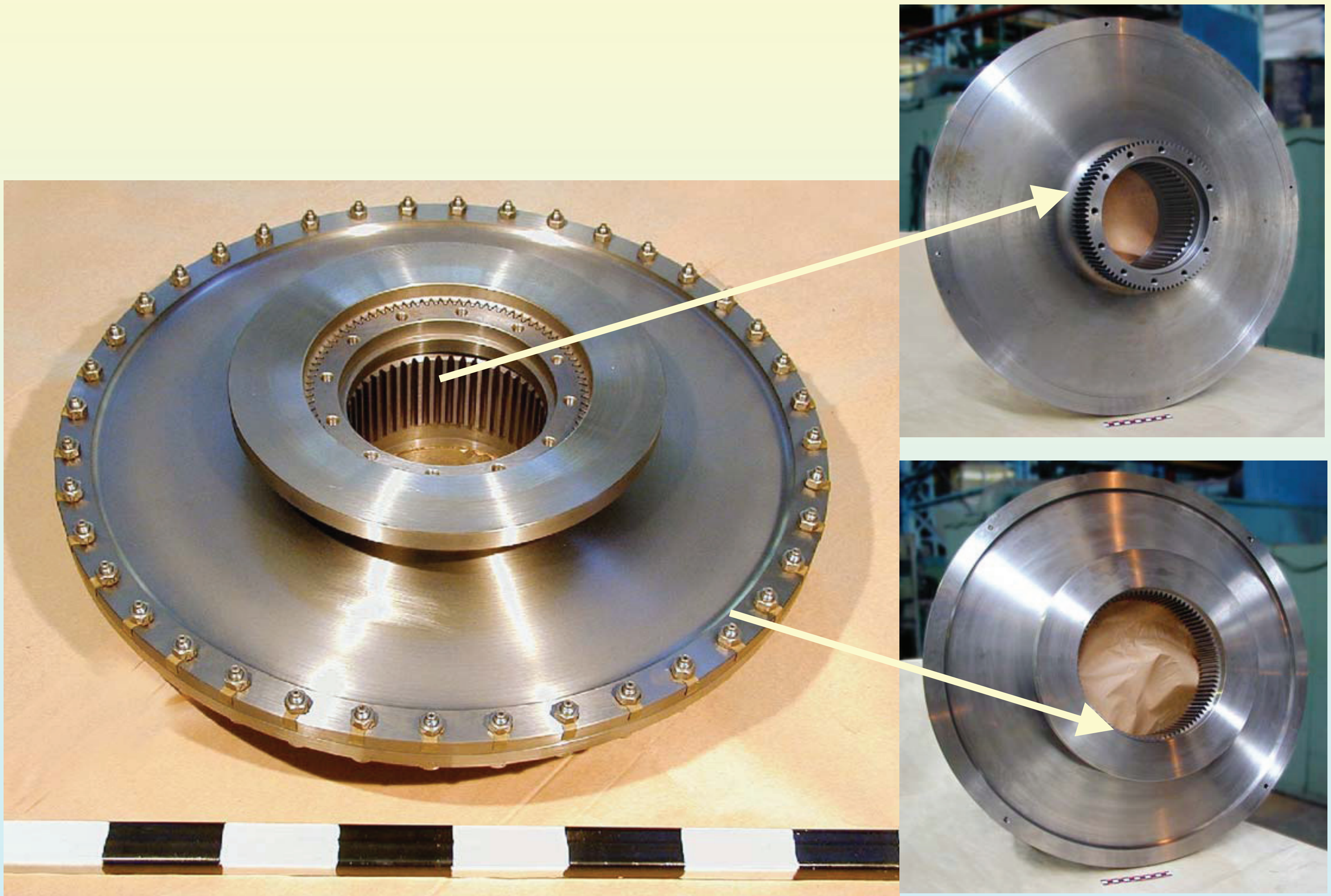
Power, kW	11
Rotation speed, s^{-1}(rpm)	100(6000)
Outer diameter of the coupling, mm	758
Number of elastic discs, pcs	1
Allowable axial displacement, mm	1
Allowable radial displacement, mm	1.3
Number of loading cycles	$1.08 \cdot 10^8$

Scaled Diaphragm Coupling Testing

ОКБМ

PCU Technology Development Program

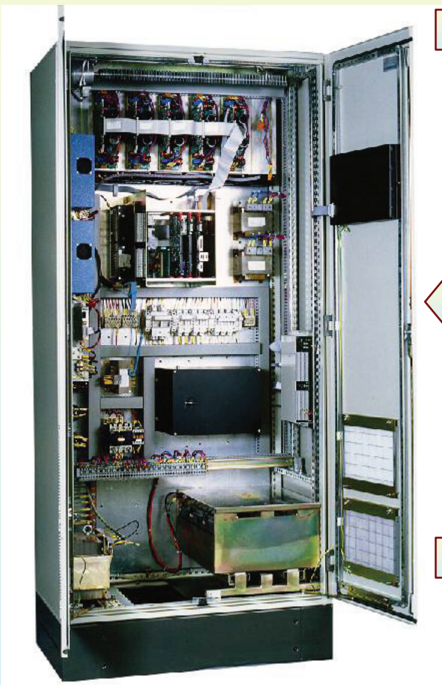
Scale Diaphragm Coupling



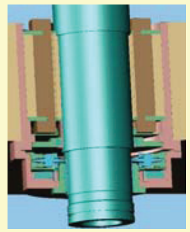
ОКБМ

PCU Technology Development Program

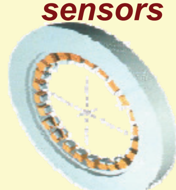
Control system



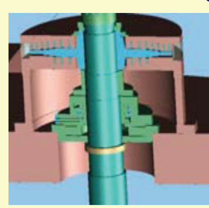
Radial EMB and catcher bearing

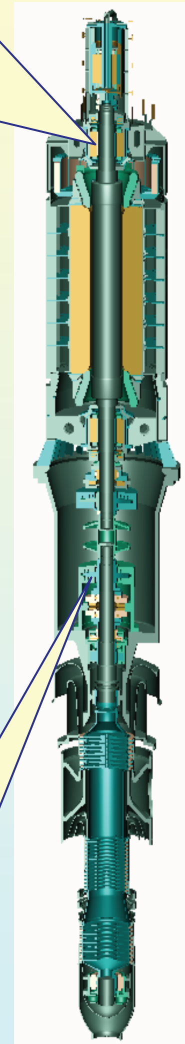


Position sensors



Axial EMB and catcher bearing





- EMB's
- CB's
- Position sensors
- Control system

Functions

Stabilization of TM rotor in central position in all operation modes

Performance characteristics

- Number of axial EMB's – 2
- Number of axial CB's – 2
- Number of radial EMB's – 4
- Number of radial CB's – 4
- Normal operating range of rotation speed 0 - 4400 rpm
- 4 critical frequencies in operating range
- Nominal load on axial EMB does not exceed 35 tons, on radial - 5 tons

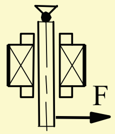
Turbomachine rotor support system

ОКБМ

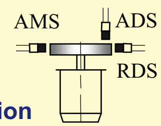
Electromagnetic Support system Development Logic

- **Minimockup**
  Optimization of EMB and Control System design in a simplified test model

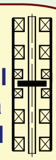
- **EMB Model**
  Verification of EMB performance characteristics



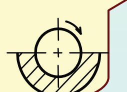
- **EMB Sensor Models**
  Verification of the performance characteristics of the EMB rotor position sensors system



- **Scaled Rotor Model**
  Verification of EMB control system performance in a multisupport scaled model of the TM rotor



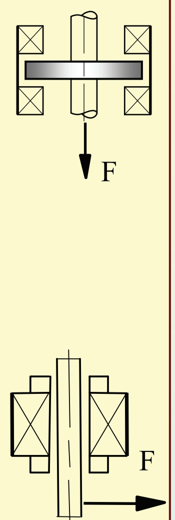
- **Friction Couple samples**
  Verification of friction characteristics of material combinations and solid lubricants in helium



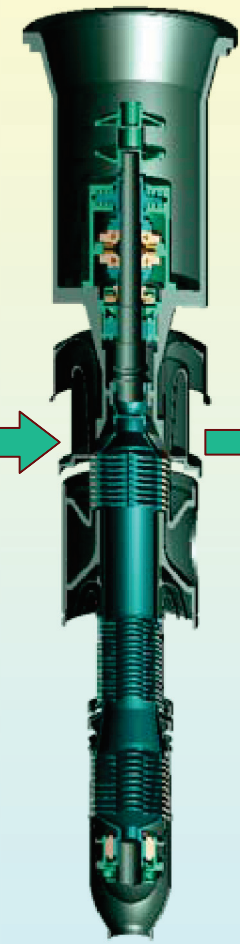
- **Catcher Bearing mockup**
  Verify operability of catcher bearings under GT-MHR conditions

- **EMB lead-outs & insulation**
  Testing of samples in helium

Development Program

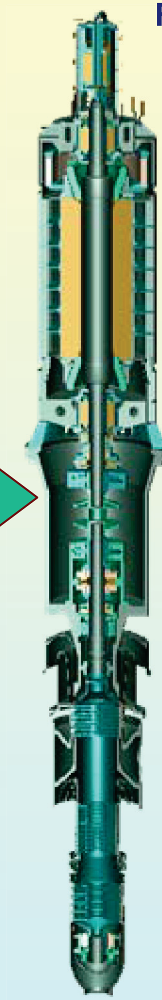


Tests of full-scale EMB and CB samples



Full-scale turbocompressor tests

First Plant



TM testing at plant site

Start-up testing at first plant

ОКБМ

PCU Technology Development Program

Minimockup

Investigation rotordynamics on full electromagnetic suspension

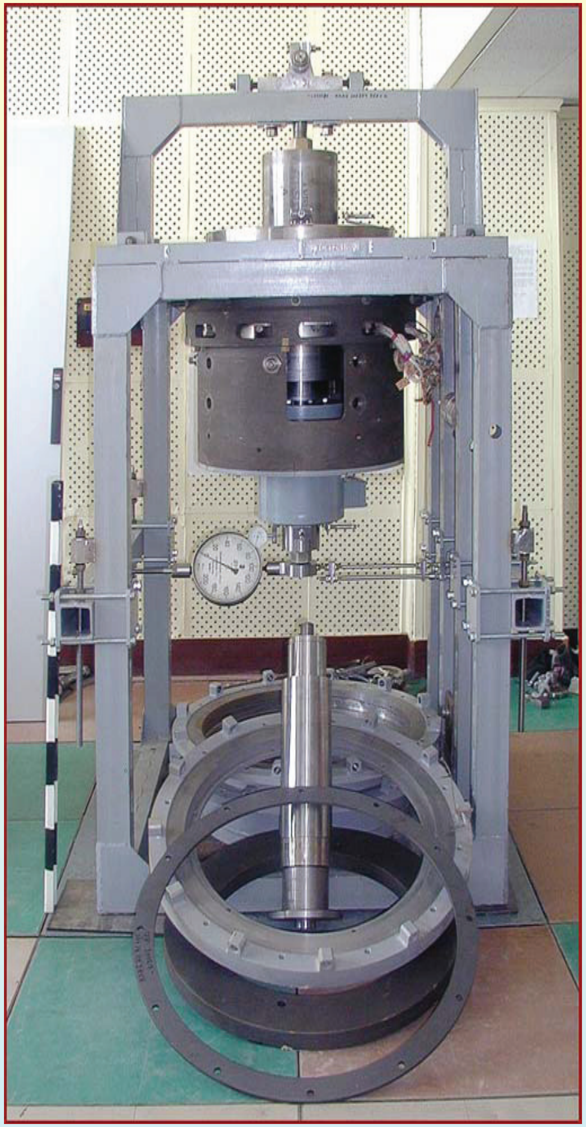
ОКБМ

PCU Technology Development Program

Investigation of position sensors

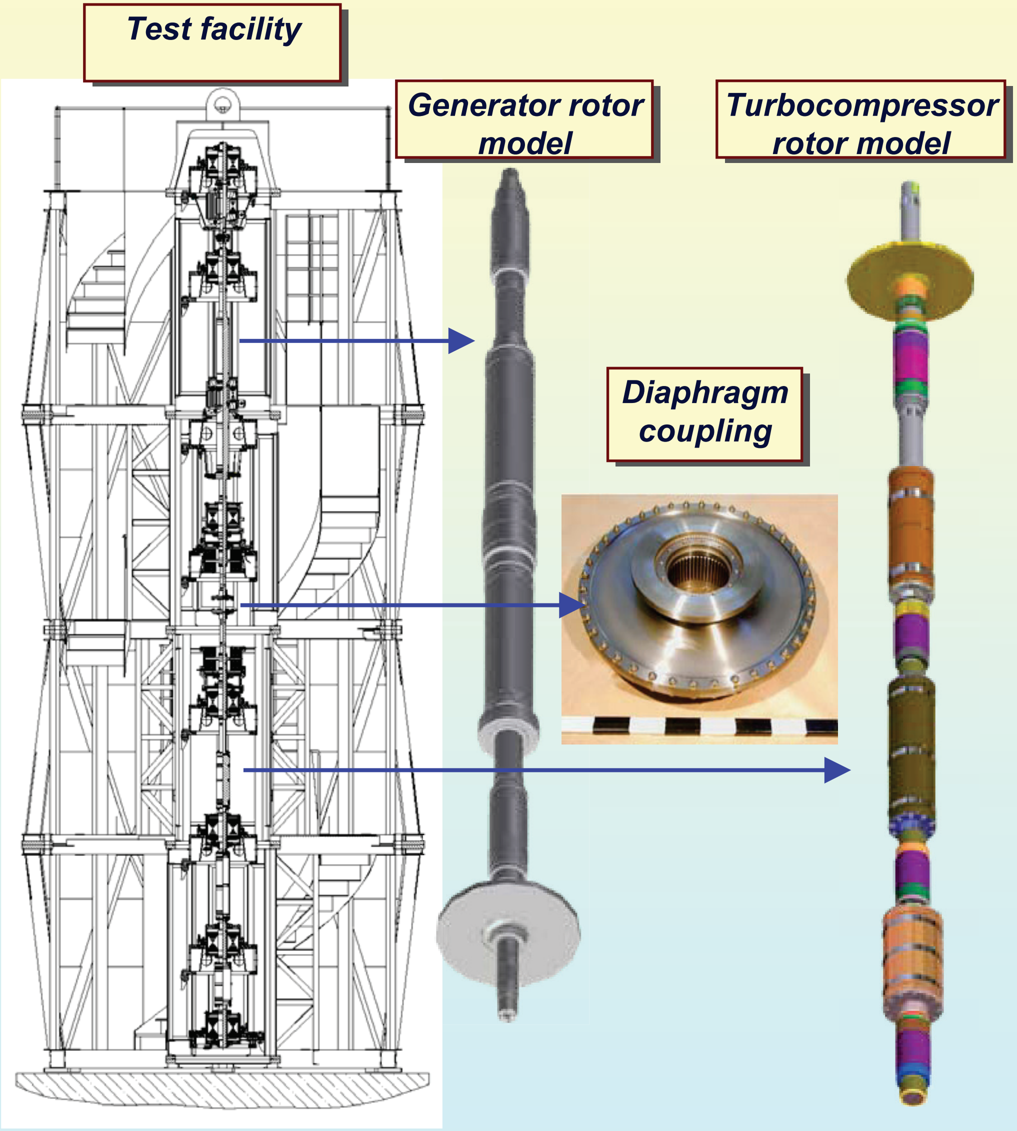
ОКБМ

PCU Technology Development Program



Investigation of radial EMB characteristics

PCU Technology Development Program



Test goal

- Verify rotordynamic calculation code
- Verify control laws and algorithms
- Experimental check of
  - critical frequency passing methods
  - ways to master external power effects
  - technology of rotor balancing in electromagnetic suspension

Performance characteristics

- Axial and radial loading devices
- Total rotor length – 10.5 m
- Total rotor weight – 1171 kg
- Operating range of rotation speed 0-6000 rpm.
- 4 critical frequencies in operating range

Status

- ✓ Design for rotor scaled model has been developed
- ✓ Test program has been developed
- ✓ Fabrication of individual components started

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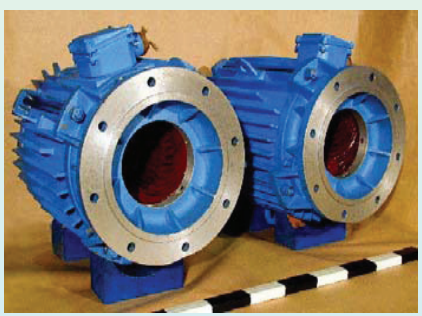
Rotor Scale Model

PCU Technology Development Program

Control boards for radial and axial EMB

Two axial EMB

Test facility electric equipment fabrication



Drives-simulators of generator and exciter

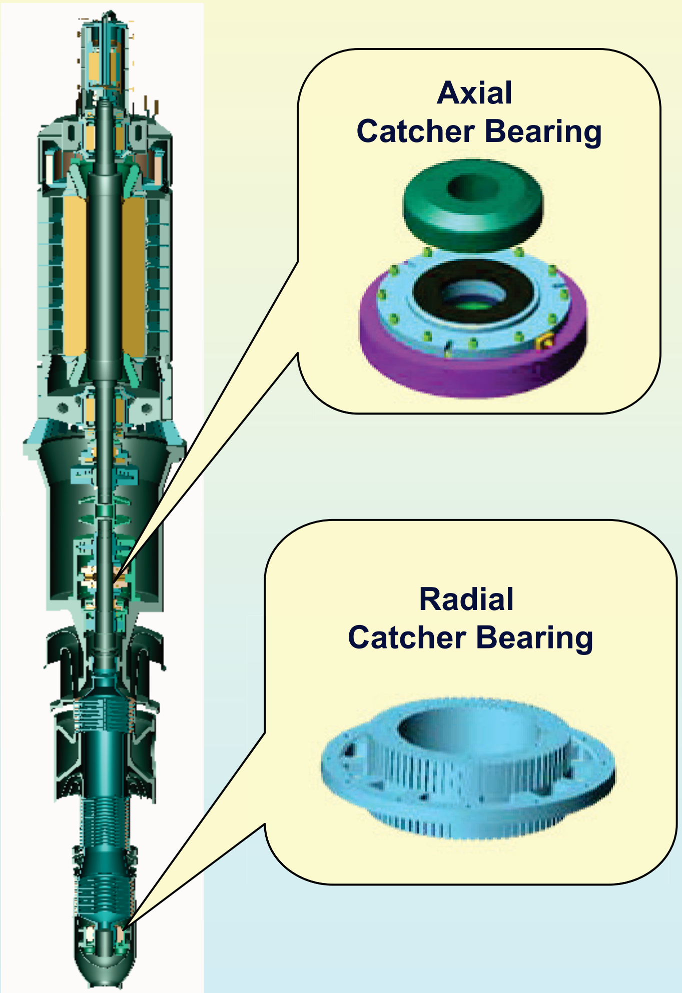
Diaphragm and rigid couplings

Coil of axial EMB

Fabricated components of Rotor Scale Model

PCU Technology Development Program

Catcher Bearing



Functions of Catcher Bearings

Catcher bearings shall provide support of the TM rotor in axial and radial directions:

- **during scheduled de-energizing of EMB of shutdown TM rotor**
- **during rotor coastdown until shutdown in case of EMB complete failure during TM operation**
- **provision of TM rotor supports during seismic loads exceeding EMB lifting capacity**

Main characteristics

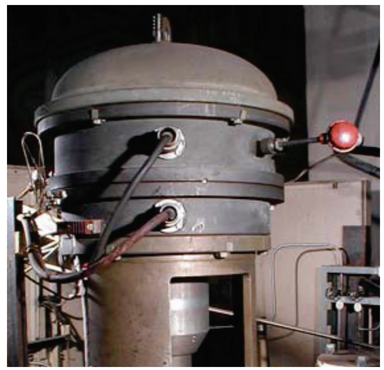
- **Number of axial CB – 2**
- **Number of radial CB – 4**
- **Maximum vertical load on axial catcher bearing – 62 tons (at EMB de-energization ration)**
- **Maximum radial load on radial catcher bearing – 28,8 tons (at MDE)**

PCU Technology Development Program

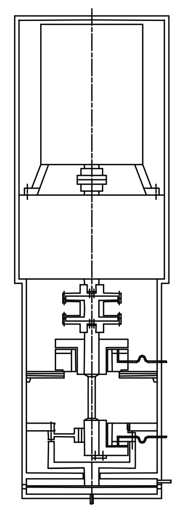
Development Program on Catcher Bearing

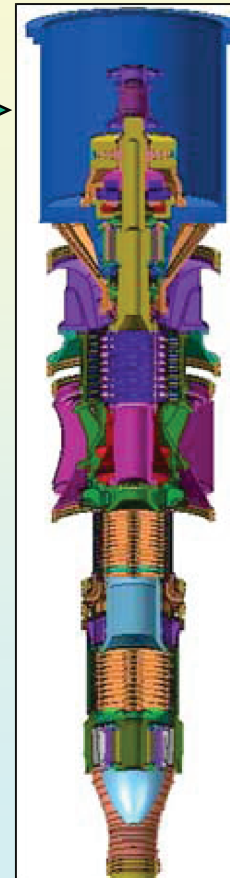
Preliminary test of friction pair materials

CB mockup test



Test of natural CB samples

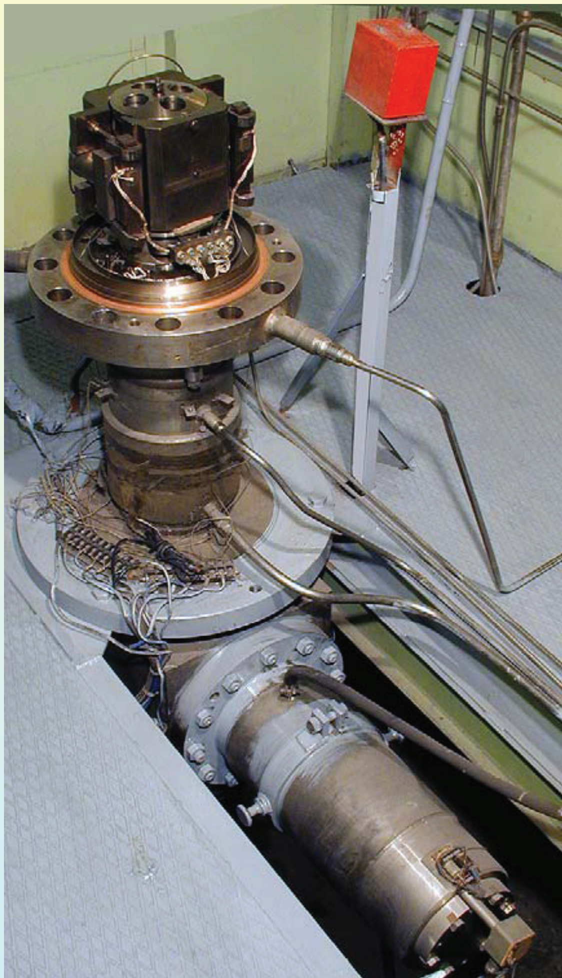


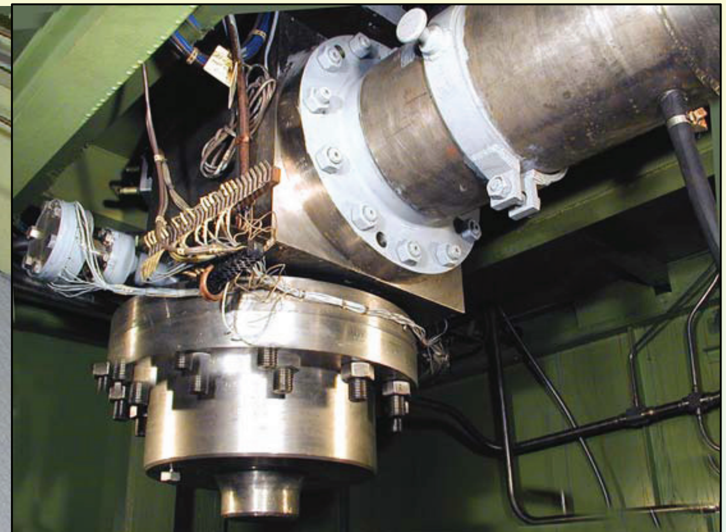


ОКБМ

PCU Technology Development Program

L-1129 test facility
Researches of materials for friction pairs of TM catcher bearings





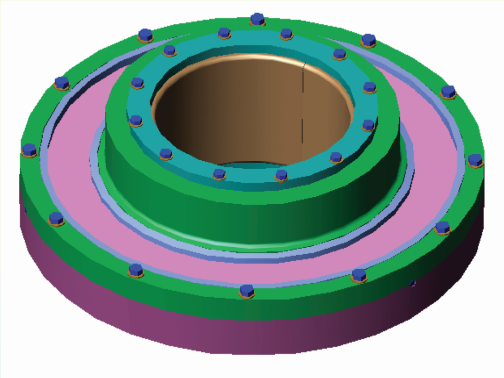
Samples of friction pairs for tests at L-1129

ОКБМ

PCU Technology Development Program

Catcher Bearing mockup

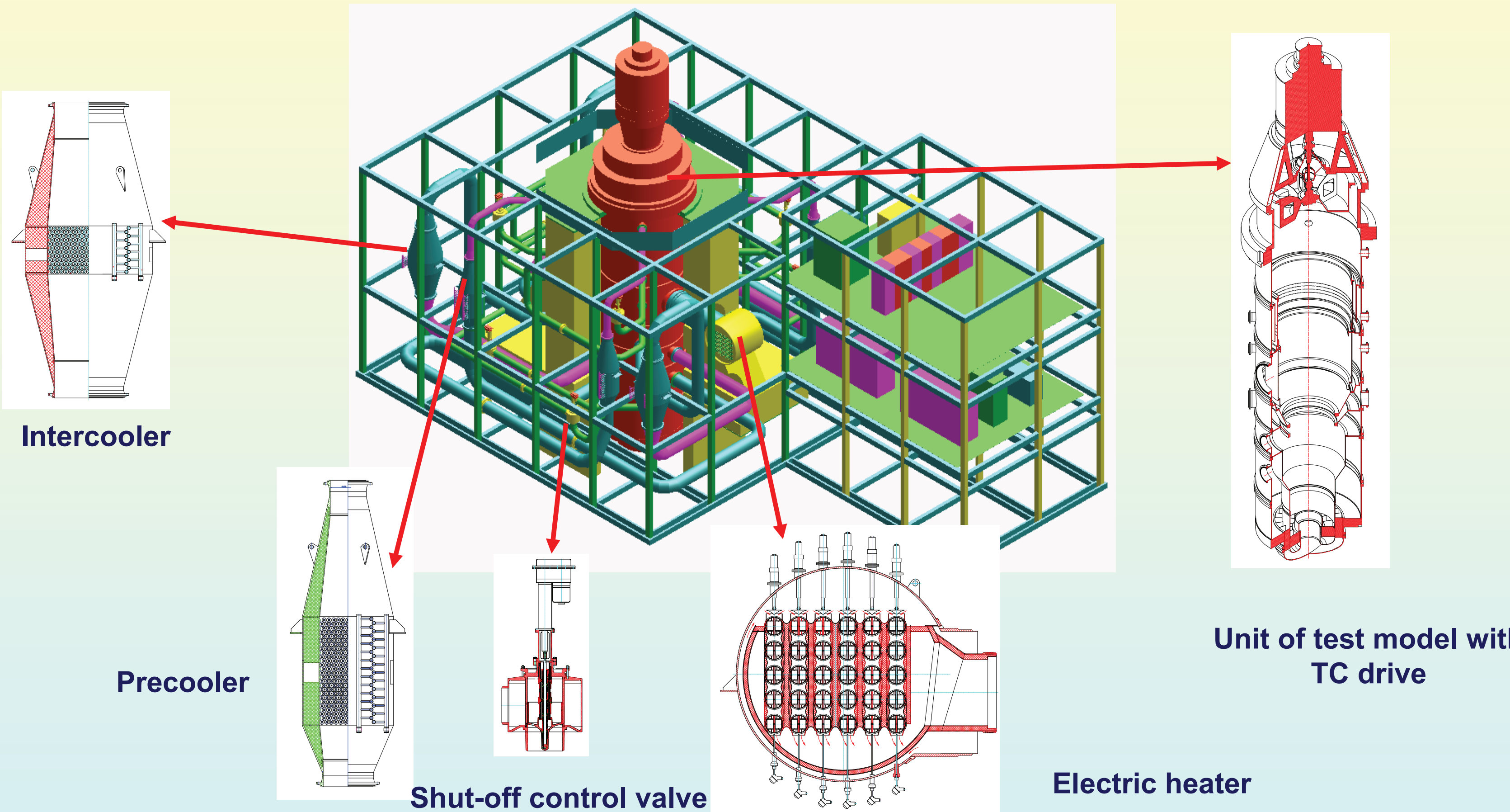
Axial catcher bearing mockup



Radial catcher bearing mockup

ОКБМ

PCU Technology Development Program



Full-scale TC Test Facility

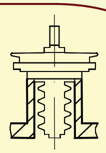
ОКБМ

PCU Technology Development Program

Development Program for Generator

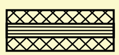
- **Lead-outs**

Testing of electric lead-outs samples of generator & EMB in helium



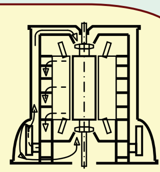
- **Generator and Exciter Stator Insulation**

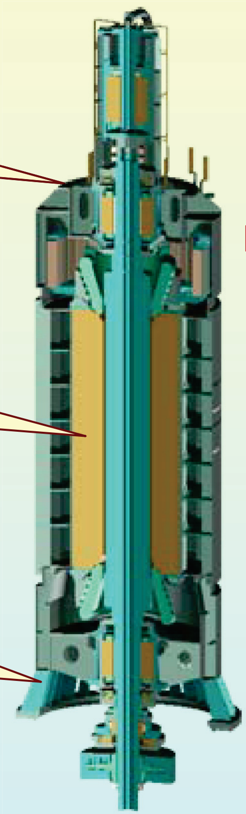
Electric insulation testing of generator & EMB in helium



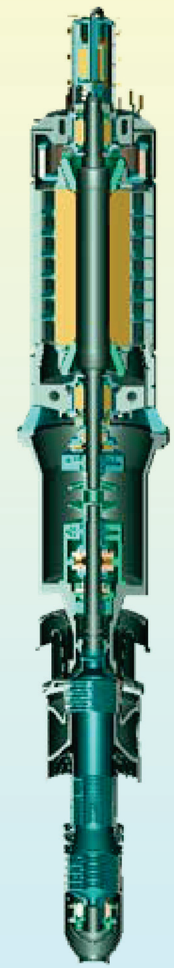
- **Generator Mockup**

Verify electric, thermal, strength computer codes





Full-scale preliminary generator tests at the end of its fabrication



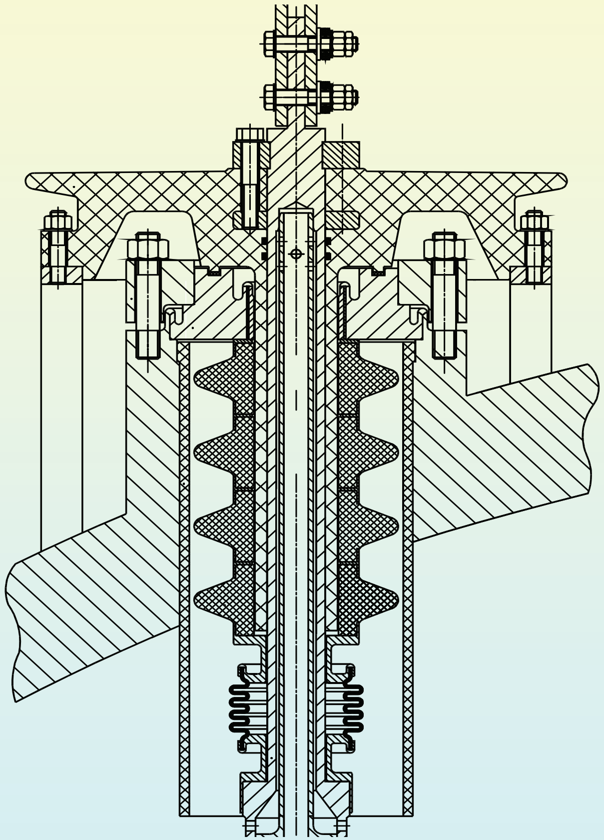
TM testing at plant site

Start-up testing at first plant

ОКБМ

PCU Technology Development Program

Insulation and Electric Lead-out Tests



Test goal

Check electrical and mechanical characteristics of electric lead-outs and insulation of generator and exciter windings in helium fluid under various pressures

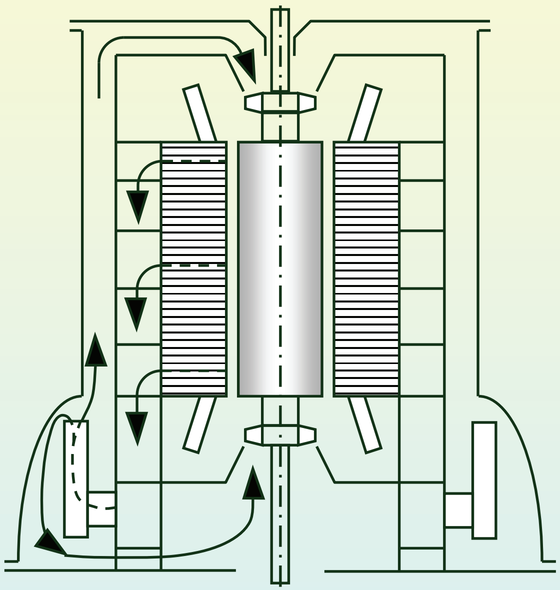
Test facility characteristic

- Operating fluid – helium
- Helium operating pressure – 10 MPa
- Helium temperature – 20…120ºC

ОКБМ

PCU Technology Development Program

Generator Mockup Tests



Test goal

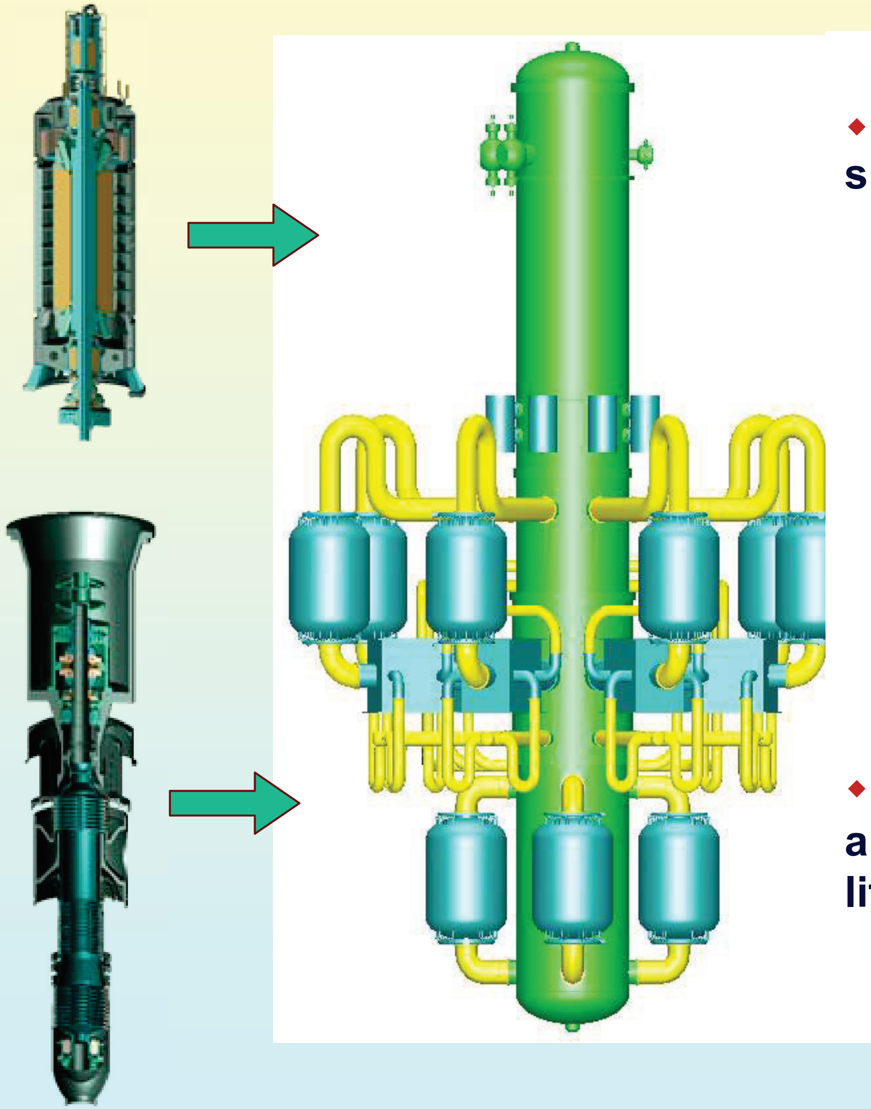
- Verify electric, thermal, strength computer codes for generator operating conditions in GT-MHR plant
- Check design solutions caused by:
  - increased rotation speed
  - vertical arrangement
  - helium cooling

Test facility characteristic

- Operating fluid – helium
- Generator mockup power – 1.5…2.5 MW

ОКБМ

TM testing at plant site



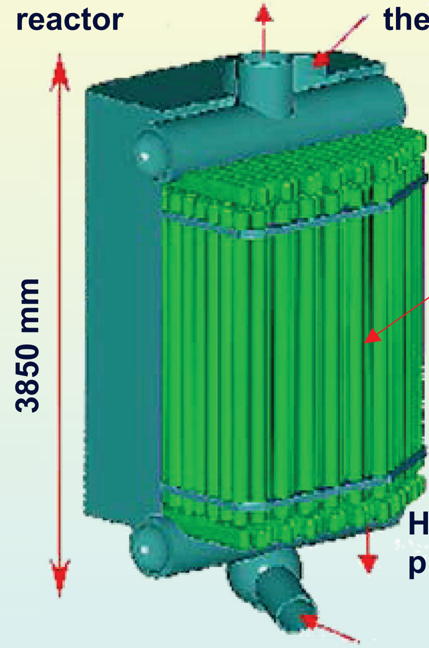
Test goal

- **Validation of pilot and commercial TM specimen characteristics:**
  - ***TM assembly quality check***
  - ***TM EMB CS operation check***
  - ***Defining generator parameters in motoring mode***
  - ***TM operation check with rotor speed ranging from 0 to 5280 rpm***
  - ***repair seal operation check***
  - ***Check of insulation, generator and TC shaft beating, shaft position relative to bearings, etc.***
- **Running-in in TM motoring mode after repair and replacement of components with expired lifetime**

ОКБМ

PCU Technology Development Program

Recuperator



Recuperator module (20 pcs.)

Recuperator mockup

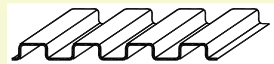
ОКБМ

PCU Technology Development Program

Development Program on Recuperator

Intensifer

Optimization of intensifier geometry

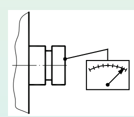


Heat Exchange Element

Thermohydraulic testing of a recuperator heat exchange element

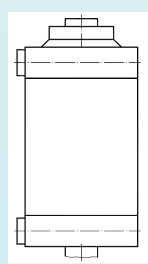
Leaky Recuperator Module Detection System Mockup

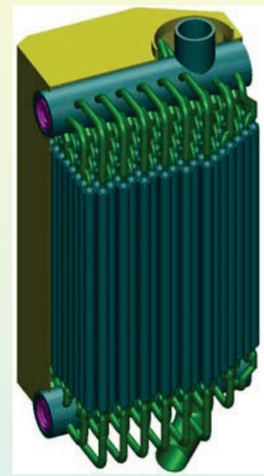
Verification of a leaky recuperator module detection system design



Recuperator Module Model

Verification of the flow distribution in the recuperator module inlet and outlet sections





Acceptance tests of a full-scale recuperator module at the end of its fabrication

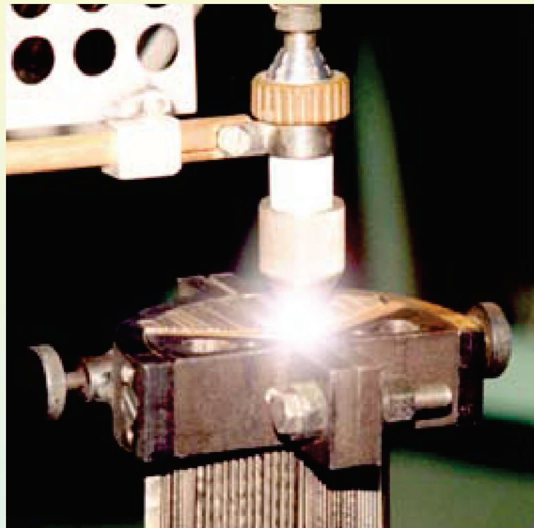
Start-up testing at first plant

ОКБМ

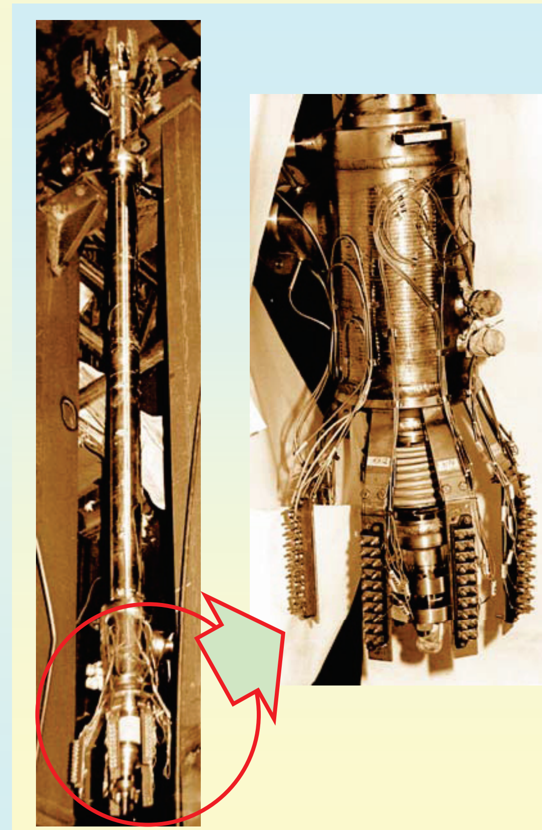
PCU Technology Development Program

RECUPERATOR

FABRICATION OF RECUPERATOR HEAT EXCHANGE ELEMENT



RECUPERATOR HEAT EXCHANGE ELEMENT



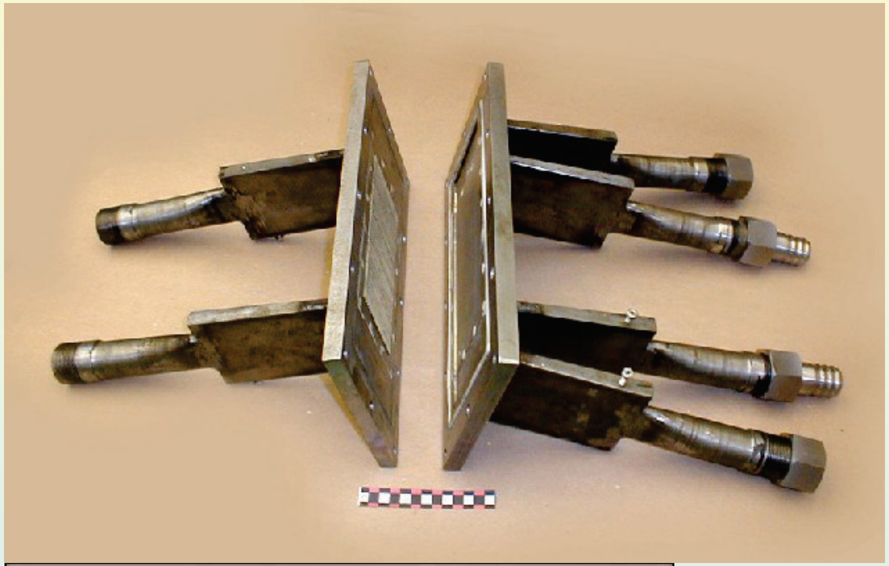
TESTS OF RECUPERATOR HEAT EXCHANGE ELEMENT AT OKBM HELIUM TEST FACILITY

ОКБМ

PCU Technology Development Program

Recuperator Intensifier Test

FLAT MODEL OF UPGRADED RECUPERATOR HEAT EXCHANGE ELEMENT

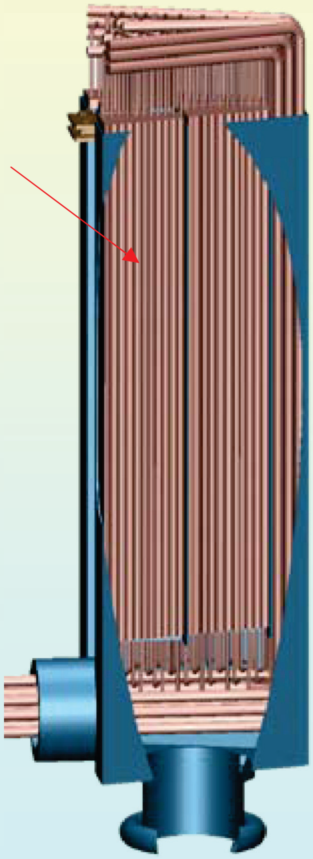


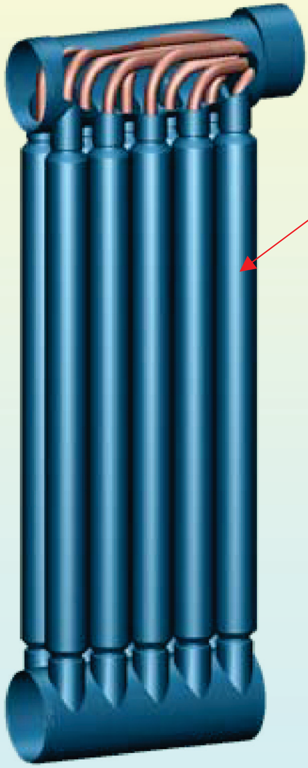
Intensifier

ОКБМ

PCU Technology Development Program

Precooler and Intercooler





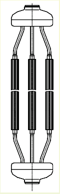
ОКБМ

PCU Technology Development Program

Development Program on Precooler / Intercooler

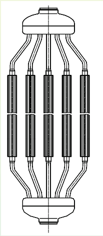
7-Tube Bundle Model

Optimization of the design of a heat exchange element in air



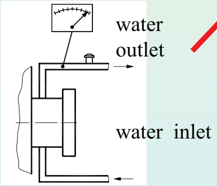
19-Tube Bundle Model

Verification of the thermal and mechanical performance of a heat exchange element in helium



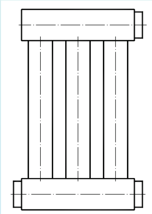
Leaky Cooler Module Detection System

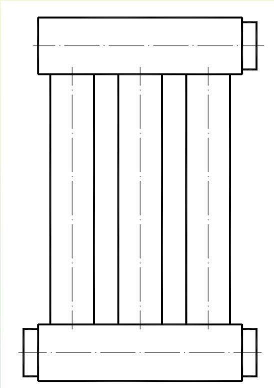
Verification of a leaky cooler detection system design



Intercooler Module Model

Verification of the flow distribution in the cooler module inlet and outlet sections





Acceptance tests of a full-scale cooler module at the end of its fabrication

Start-up testing at first plant

ОКБМ

PCU Technology Development Program

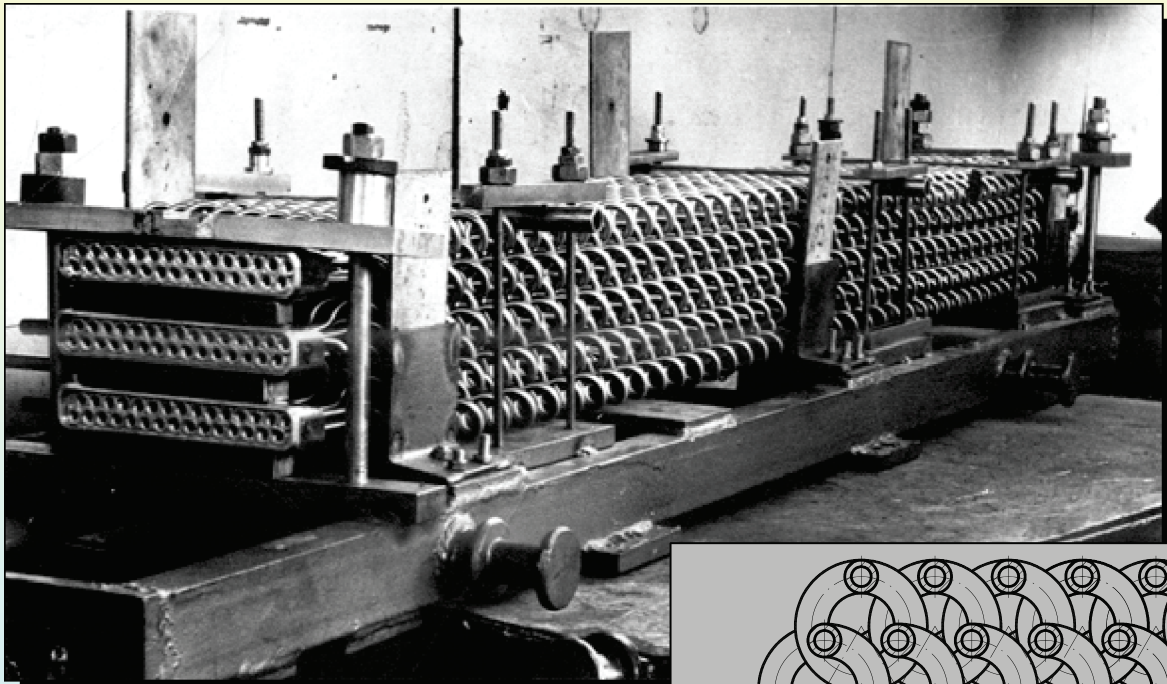
COOLER MODELS FOR EXPERIMENTAL INVESTIGATIONS OF FINNED HEAT-EXCHANGE SURFACES

ON THE LEFT – READY-MADE MODEL;
ON THE RIGHT – SECOND MODEL FRAGMENTS

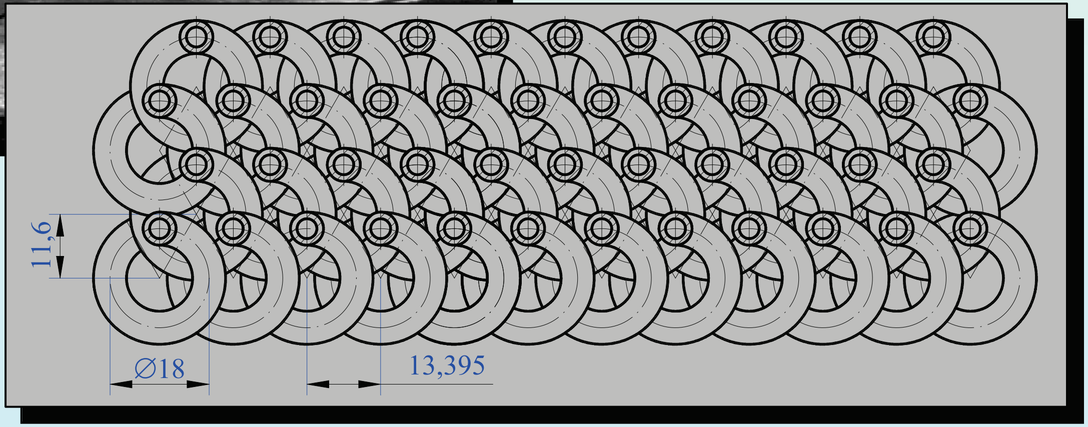
ОКБМ

PCU Technology Development Program

Generator Gas Cooler (Backup for Precooler/Intercooler)



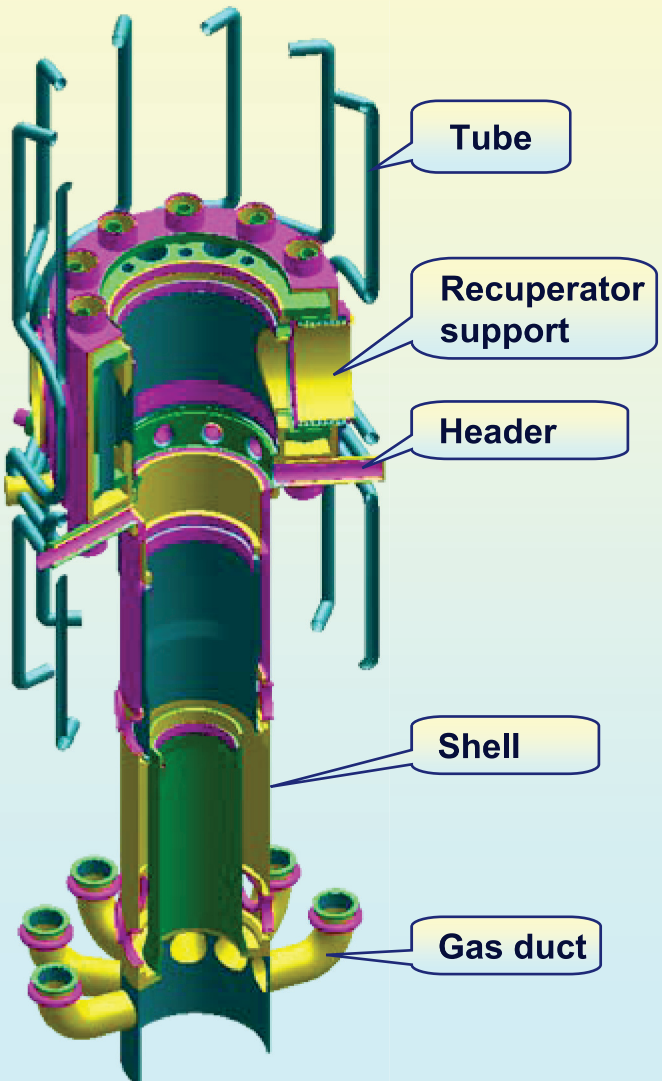
Nested Small Diameter Helical Coil Heat Exchanger Concept



ОКБМ

PCU Technology Development Program

In-vessel metalworks



MAIN FUNCTIONS

- *Provide attachment interface and load path between PCU components and vessel*
- *Create helium circulation path and restrict in-circuit leaks*
- *Restrict heat exchange between helium flows with different temperatures*

MAIN TECHNICAL CHARACTERISTICS

- *Material - plates, sheets and tubes from steels*
  - *10Cr9MoVNb*
  - *CrNi55MoWZr*
  - *08Cr18Ni10Ti*
- *Mass, t - 270*

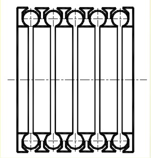
ОКБМ

PCU Technology Development Program

IN-VESSEL STRUCTURES (IVM) AND BYPASS GAS MIXER DEVELOPMENT PLAN

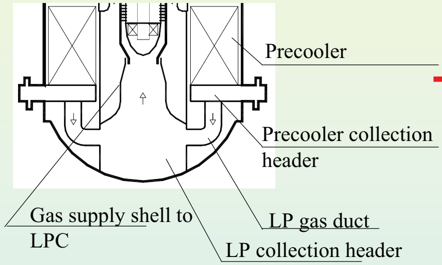
Expansion Joint

Verification of expansion joint design



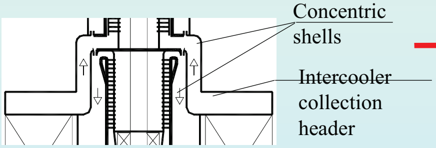
Flow Path (Precooler Outlet - LPC Inlet)

Verification of flow distribution between the precooler outlet and the low pressure compressor inlet

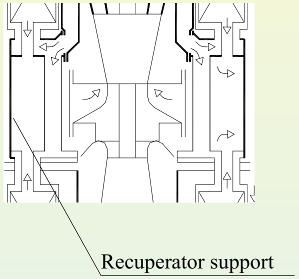


Flow Path (Intercooler Outlet - HPC Inlet)

Verification of flow distribution between the intercooler outlet and the high pressure compressor inlet

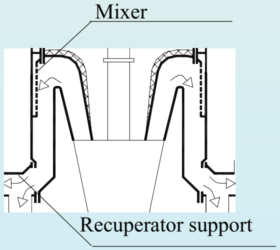


Inlet Flow Path from Turbine Outlet to Recuperator Modules Inlet



Verification of flow distribution between turbine outlet and the low-pressure side recuperator modules

Flow Path (TM Outlet Chamber - Recuperator Support Inlet)



Study of mixing process of bypass flow with turbine outlet flow

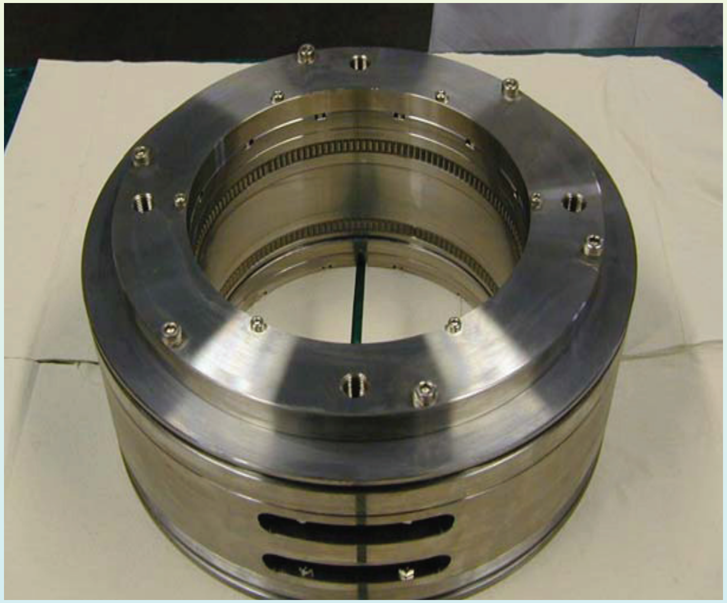
Start-up testing at first plant

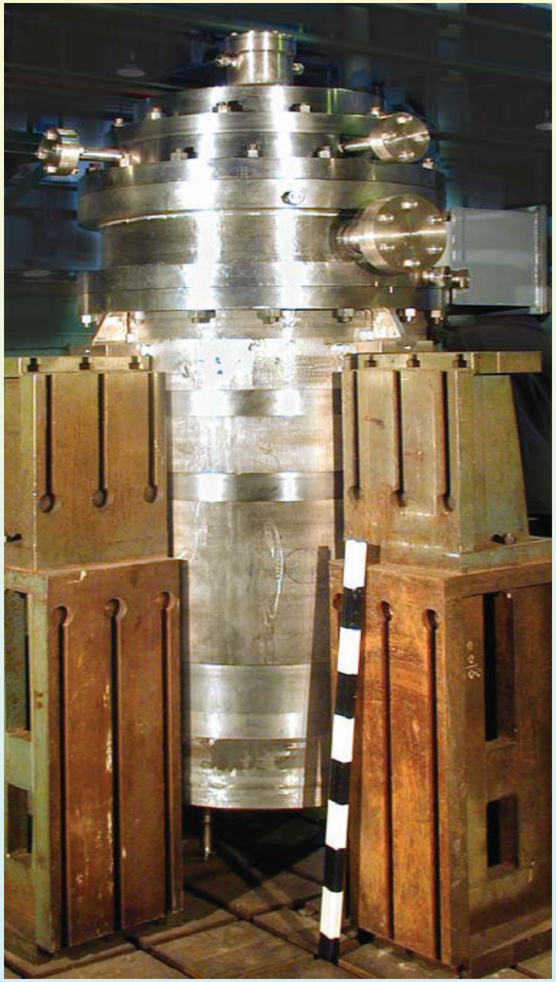
ОКБМ

PCU Technology Development Program

Dry Gas Rotor Seal

Pilot sample of dry gas seal of EKK EAGLE INDUSTRY CO., LTD Japanese company





Running part of the test facility was fabricated to test various seal options

ОКБМ

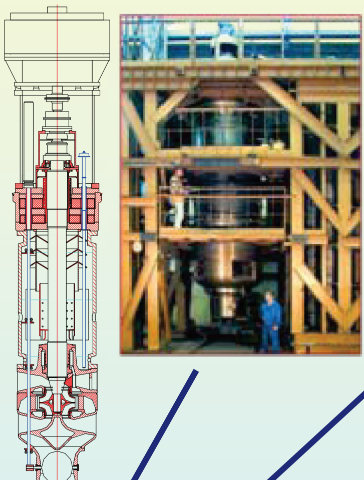
Part II.
PCU Experience

ОКБМ

Experience of Vertical Machine Creation

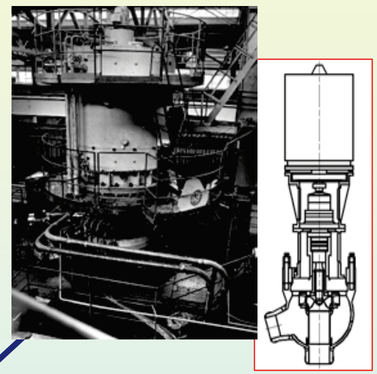
TsNN-9 electric pump (OKBM, Russia)

- Power – 3.5 MW
- Rotation speed – 1000 rpm
- Rotor mass – 8.2 t
- Rotor length – 12 m
- Unit mass – 120 t



TsVN-8 electric pump (OKBM, Russia)

- Power – 4.3 MW
- Rotation speed – 1000 rpm
- Rotor mass – 3 t
- Rotor length – 7.8 m
- Unit mass – 106 t

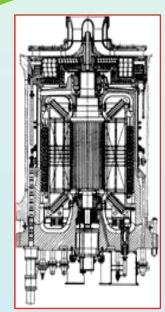


SV 865/232-28UHL4 hydrogenerator (ELSIB, Russia)

- Power – 200 MW
- Rotation speed – 214.3 rpm
- Unit mass– 1031 t

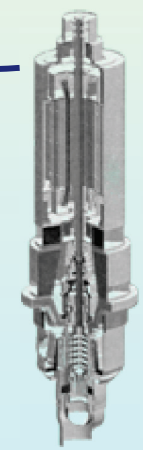
GT-MHR turbomachine (design)

- Power – 287 MW
- Rotation speed – 4400 rpm
- Rotor length – 26 m
- Rotor generator mass – 35 t
- Rotor TC mass – 30 t
- Mass – 500 t



EG-90/1.25 gas circulator (OKBM, Russia)

- Power – 6.3 MW
- Rotation speed – 6000 rpm
- Rotor mass – 4.1 t
- Rotor length – 3.5 m
- Mass – 55 t



PBMR turbogenerator (design)

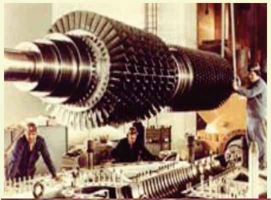
- Power – 180 MW
- Rotation speed – 3000 rpm
- Rotor mass – 88 t

0 100 200 300 MW

ОКБМ

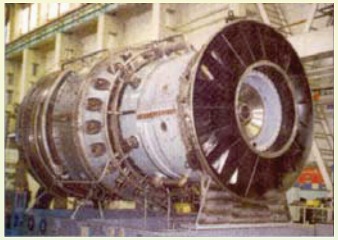
Experience of Turbine and Compressor Creation

EVO gas-turbine plant (Oberhausen, Germany)



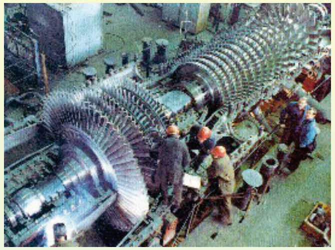
- Power – 50 MW
- Rotation speed – 3000 rpm
- Turbine:
  - Number of turbine stages – 7x11
  - Maximum diameter – 1500 mm
- Compressor:
  - Number of compressor stages – 10x15
  - Maximum diameter 1000 mm

GTE-110 gas-turbine plant (Rybinskiye motory, Russia)



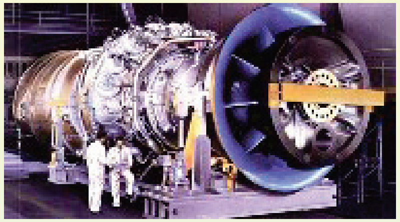
- Power – 110 MW
- Rotation speed – 3000 rpm
- Number of turbine stages – 4
- Number of compressor stages – 15
- TC length – 7 m
- TC diameter – 3,1 m
- TC mass – 50 t

GTE-150 gas-turbine plant (LMZ, Russia)



- Power – 150 MW
- Rotation speed – 3000 rpm
- Turbine:
  - Number of turbine stages – 4
  - Maximum diameter – 2800 mm
- Compressor:
  - Number of compressor stages – 14
  - Maximum diameter 2100 mm
- TC length – 15 m
- TC diameter – 5 m
- TC mass – 220 t

GE MS 9001F gas-turbine plant , USA



- Power – 226 MW
- Rotation speed – 3000 rpm
- Turbine:
  - Number of turbine stages – 3
  - Maximum diameter – 3251 mm
- Compressor:
  - Number of compressor stages – 18
  - Maximum diameter 2515 mm
- TC length – 14,5 m
- TC diameter – 4,8 m
- TC mass – 300 t

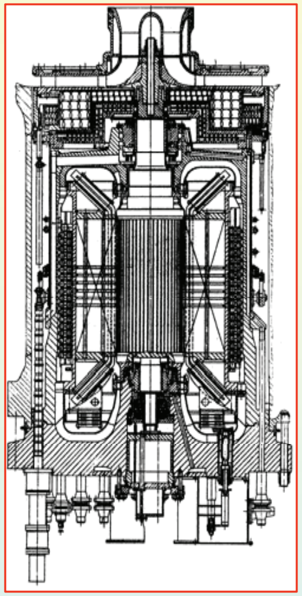
GT-MHR turbocompressor (design)

- Power – 300 MW
- Rotation speed – 4400 rpm
- Turbine:
  - Number of turbine stages – 10
  - Maximum diameter – 1490 mm
- Compressor:
  - Number of compressor stages – 11x14
  - Maximum diameter – 1565 mm
- TC length – 13,5 m
- TC mass – 180 t

ОКБМ

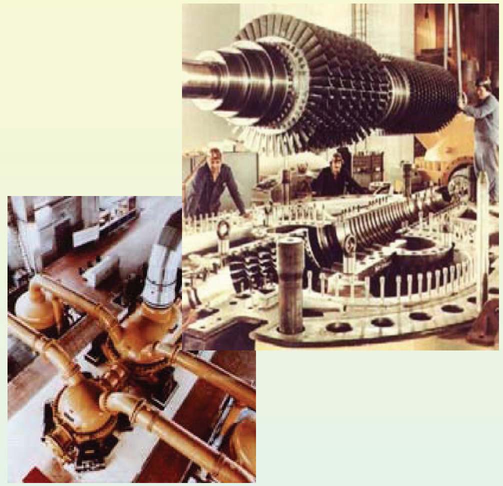
Experience of Helium Machine Creation

EG-90/1.25 gas circulator (OKBM, Russia)



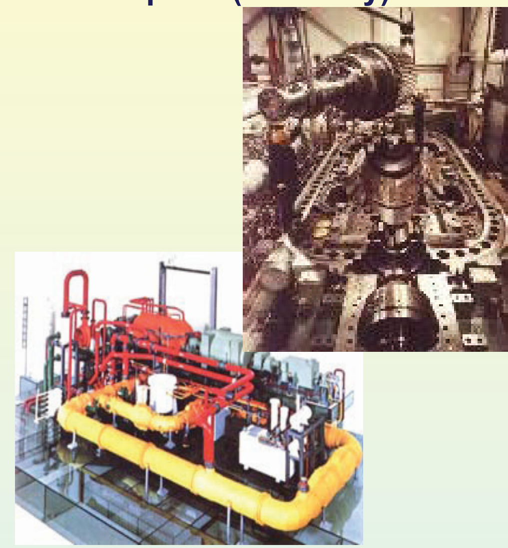
- Power – 6.3 MW
- Rotation speed – 6000 rpm
- Rotor mass – 4.1 t
- Rotor length – 3.5 m
- Mass – 55 t

EVO gas-turbine plant (Oberhausen)



- Power – 50 MW
- Rotation speed – 3000 rpm
- Turbine:
  - Number of stages – 7x11
  - Maximum diameter – 1500 mm
- Compressor:
  - Number of stages – 10x15
  - Maximum diameter – 1000 mm

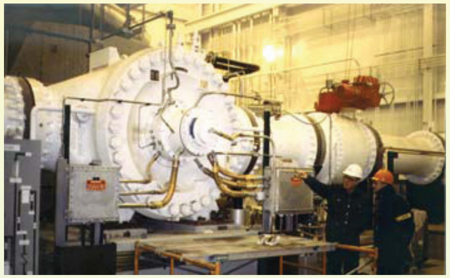
HHV test plant (Germany)



- Power – 90 MW
- Rotation speed – 3000 rpm

ОКБМ

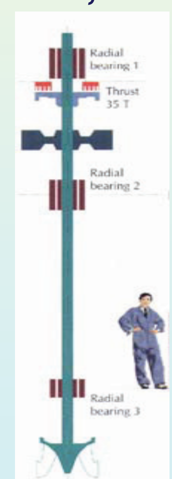
EMB experience



GT-MHR TURBOMACHINE
300 MW / 4400 rpm
Up to 35 tons – on axial EMB,
Up to 6 tons - on radial EMB

S2M, COMPRESSOR
29 MW / 5700 rpm, up to 6 tons on radial EMB

S2M, Hydrogenerator,
300 rpm,
Up to 35 tons – on axial EMB, rotor height - 7m



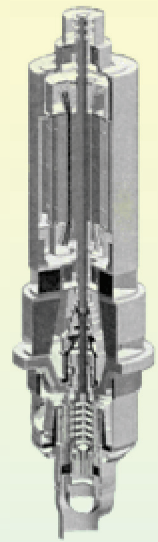
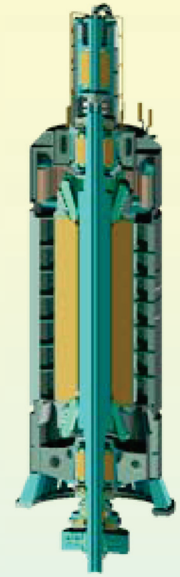
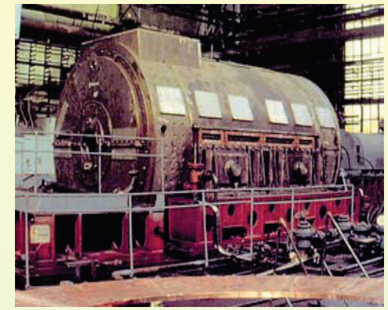
Waukesha Bearing,
Electric compressor.
Up to 8 tons – on axial and radial EMB

Load on EMB, tons

0 10 20 30 Load on EMB, tons

ОКБМ

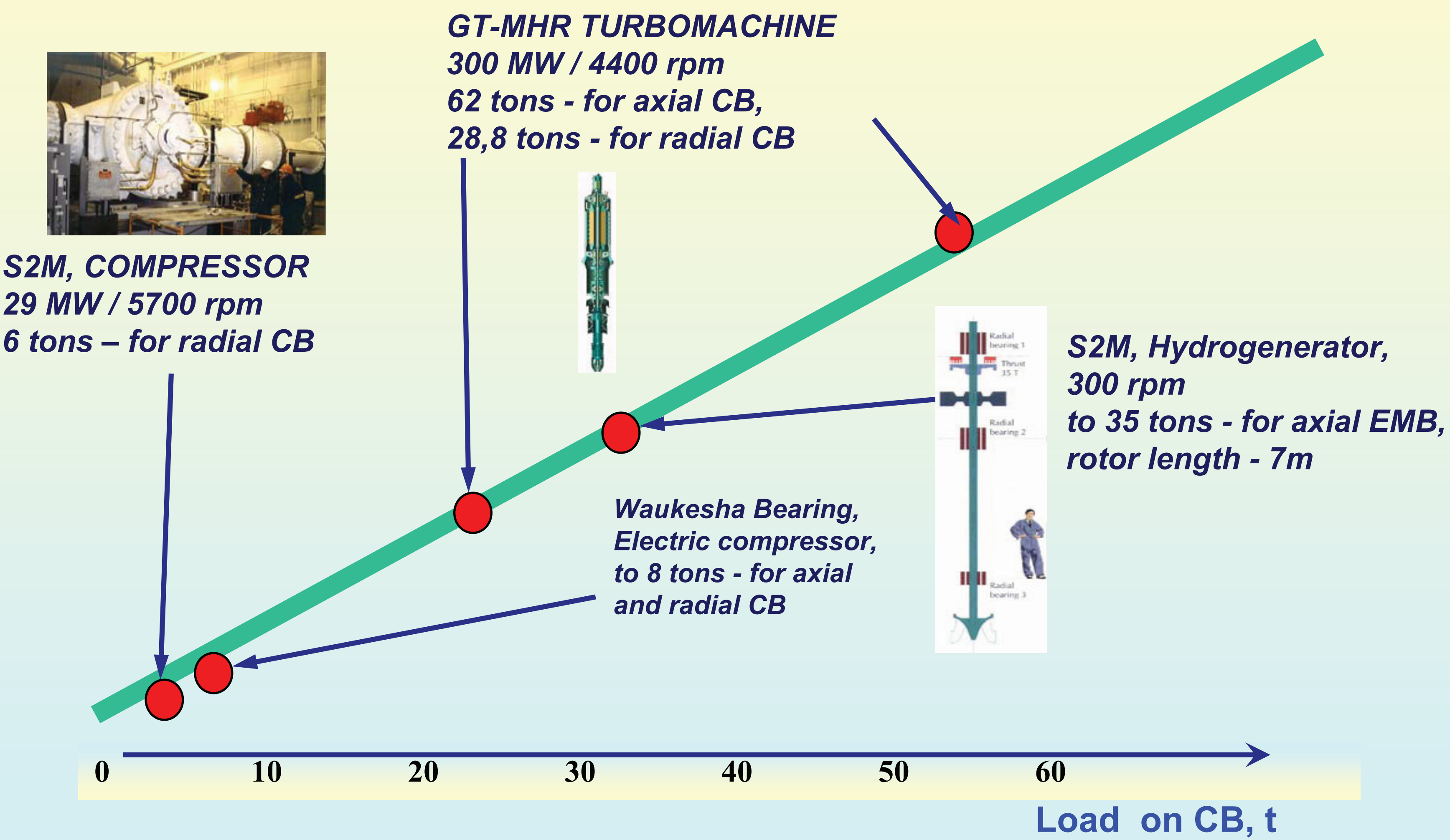
Generator Creation Experience

	Hydrogenerator SV 865/232-28УХЛ4, ELSIB, Russia	*PBMR plant generator, design*	*GT-MHR plant generator, design*	*Generator ТВВ-320-2ЕУ3, Elektrosila, Russia*	*Generator Т3В-1500-2У3, Elektrosila, Russia*
Power, MW	200	180	287	320	1500
Voltage, kV	15.75	12.5	20	20	27
Rotation speed, rpm	214.3	3000	4400	3000	3000
Efficiency, %	98.4	96	97.7	98.75	99
Mass, m	1031		275	257	615

ОКБМ

Experience in Creation of Catcher Bearings



GENERAL ATOMICS

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